UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

July 19, 2006

R. T. Ridenoure, Vice President
Omaha Public Power District
Fort Calhoun Station FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: INSPECTION REPORT 050-00285/06-013; 072-00054/06-002

Dear Mr. Ridenoure:

Between March 13 and May 5, 2006, the NRC conducted an inspection of your dry fuel storage preparation activities. This inspection involved site visits by inspectors on four separate occasions. The inspection was performed to evaluate your implementation of the requirements for dry fuel storage contained in 10 CFR Part 72 and in the NUHOMS Certificate Of Compliance, Final Safety Analysis Report, and NRC Safety Evaluation Report. The enclosed inspection report presents the results of the inspection, which were discussed with members of your staff at the conclusion of each site visit and during the final exit briefing held by telephone on June 13, 2006.

The inspection reviewed the auxiliary building crane and crane support structure, the 10 CFR 72.48 evaluation for use of the new lightweight OS197L transfer cask, the Fort Calhoun Station 10 CFR Part 50 programs related to dry fuel storage and the heavy loads testing program. As a result of the 10 CFR 72.48 review, numerous telephonic communications were held between your staff and NRC staff and management to discuss several aspects of the planned cask loading operations and whether they could be performed pursuant to 10 CFR 72.48 without prior NRC approval. In addition, these matters were discussed in an open meeting held in NRC Headquarters on May 24, 2006. As a result of these discussions, Omaha Public Power District concluded that an exemption request would be submitted to the NRC to seek approval to use the new transfer cask at the Fort Calhoun Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Should you have any questions concerning this inspection, please contact the undersigned at (817) 860-8191 or Mr. Scott Atwater at (817) 860-8286.

Sincerely,

D. Blair Spitzberg, Ph.D., Chief
Fuel Cycle and Decommissioning Branch

Docket Nos.: 50-285
72-054
License No.: DPR-40

Enclosure:
NRC Inspection Report 050-00285/06-013; 072-00054/06-002

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	050-00285; 072-00054
License:	DPR-40
Report No:	050-00285/06-013; 072-00054/06-002
Licensee:	Omaha Public Power District
Facility:	Fort Calhoun Station
Location:	P.O. Box 550 Fort Calhoun, NE 68023-0550
Dates:	March 13 through May 5, 2006
Inspectors:	S.P. Atwater, Health Physicist, Region IV/DNMS R.L. Kellar, P.E., Health Physicist, Region IV/DNMS L.M. Willoughby, Resident Inspector, Fort Calhoun E.M. Garcia, Health Physicist, Region IV/DNMS J.P. Adams, Reactor Inspector, Region IV/DRS R.R. Temps, Senior Safety Inspector, SFPO/TSSI
Accompanied By:	J.M. Sebrosky, Senior Project Manager, SFPO
Licensing Basis Research Provided By	J. Solis, Thermal Engineer, SFPO/TRD/CSHT S.R. Helton, Nuclear Engineer, SFPO/TRD/CSHT E.A Thompson, Health Physicist, SFPO/TRD/CSHT J.S Ma, Structural Engineer, NRR/DE/EGCB S.K. Samaddar, Structural Engineer, NRR/DE/EGCB
Approved By:	D.B. Spitzberg, Ph.D., Chief Fuel Cycle and Decommissioning Branch
Attachments:	1. Supplemental Information 2. Inspector Notes

EXECUTIVE SUMMARY

Fort Calhoun Station
NRC Inspection Report 050-00285/06-013; 072-00054/06-002

The Omaha Public Power District (OPPD) had selected the Standardized NUHOMS Horizontal Modular Storage System for dry storage of spent nuclear fuel at the Fort Calhoun Station. The Nuclear Regulatory Commission (NRC) had certified the NUHOMS cask system for storage of irradiated fuel under Certificate of Compliance No. 72-1004.

On December 5, 2005 the Nuclear Regulatory Commission (NRC) approved Amendment 8 to Certificate of Compliance (CoC) No. 72-1004. Amendment 8, Attachment A contained the Technical Specifications for the 32PT canister system, which had been selected for use at the Fort Calhoun Station. The Technical Specifications ensured that the Independent Spent Fuel Storage Facility (ISFSI) would be operated within its design limits and in compliance with the requirements of 10 CFR Part 72.

The NRC inspection included four onsite visits by inspectors between March 13 and May 5, 2006 to evaluate the licensee's implementation of the Amendment 8 Technical Specifications and 10 CFR Part 72 requirements. The onsite inspection effort included the following activities:

1) The auxiliary building crane and crane support structure which will be used to move the transfer cask loaded with irradiated fuel were inspected on March 14-16, 2006 as part of the heavy loads program. Inspectors from the Region IV Division of Nuclear Materials Safety (DNMS) and the Region IV Division of Reactor Safety (DRS) conducted this inspection to evaluate licensee compliance with the crane licensing basis documents. Support was provided by the Office of Nuclear Reactor Regulation (NRR) to evaluate the adequacy of the auxiliary building structure to support the crane during a seismic event.

2) The 10 CFR 72.48 evaluation under which Transnuclear introduced their new lightweight OS197L transfer cask was inspected on April 3-6, 2006 at the Fort Calhoun Station and on April 19, 2006 at the Transnuclear headquarters. Inspectors from Region IV DNMS and the Spent Fuel Project Office (SFPO) evaluated the thermal and shielding performance of the new OS197L transfer cask and questioned the ability of the OS197L transfer cask to meet three of the Certificate of Compliance Technical Specifications. Due to time constraints, the licensee elected to request an exemption from the Technical Specifications pursuant to 10 CFR 72.7 rather than engage in further study to definitively answer the questions raised. Transnuclear may be subject to future inspections in order to determine the ability of the OS197L transfer cask to meet the Technical Specifications.

3) The Fort Calhoun Station 10 CFR Part 50 programs related to dry fuel storage were evaluated on April 10-13, 2006 to determine their adequacy for dry fuel storage operations. Inspectors from Region IV DNMS and from SFPO conducted this inspection.

4) Heavy loads testing was observed on May 1-5, 2006 as part of the pre-operational testing program. An inspector from Region IV DNMS and the Fort Calhoun Station resident inspector conducted this inspection.

The following provides a summary of the results of the inspection. Details are provided in the Inspector Notes contained in Attachment 2 to this report.

Auxiliary Building Crane

- The auxiliary building 75 ton Ederer crane was approved by the NRC as single-failure-proof. The crane design and construction met the seismic criteria contained in NUREG 0554 (Attachment 2, Crane Design Basis).

- The auxiliary building structure supporting the crane met the seismic criteria for Class I structures contained in the Fort Calhoun Station Updated Safety Analysis Report (Attachment 2, Crane Design Basis).

- The hoist design features of the auxiliary building crane, as described in the Ederer topical report were intact and operable. The features included the drum support structure, hoist holding brakes, provisions for manual operation, and minimum wire rope breaking strength (Attachment 2, Crane Hoist Design).

- The auxiliary building crane was inspected, tested and maintained in accordance with the ASME Code, NUREG 0554, and the crane manufacturer's instructions (Attachment 2, Crane Inspection, Load Testing and Maintenance).

- The crane safety systems were inspected and tested in accordance with the crane manufacturer's instructions and the Fort Calhoun Updated Safety Analysis Report (Attachment 2, Crane Safety Systems).

Emergency Planning and Fire Protection

- The Emergency Plan and Fire Protection Plan were expanded to envelop the ISFSI and site personnel had been trained on the changes. Offsite emergency vehicles and services were provided direct access to the ISFSI (Attachment 2, Emergency Planning, Fire Protection).

Fuel Selection and Verification

- The spent fuel assemblies selected for the planned loading campaign of ten canisters met the Technical Specification requirements for integrity, physical characteristics, enrichment, burnup, cooling time and decay heat load. A loading plan had been developed and approved for each of the ten canisters. The minimum spent fuel pool boron concentration required by Technical Specifications had been established. The NRC exemption will limit the licensee's actual loading campaign to four canisters containing spent fuel assemblies with lower decay heat values and greater cooling times. These assemblies will be verified to meet Technical Specifications and the NRC exemption requirements prior to loading (Attachment 2, Fuel Selection/Verification).

General License Conditions

- The 10 CFR 72.48 inspection questioned the ability of the OS197L transfer cask to meet three of the Certificate of Compliance Technical Specifications. Due to time constraints, the licensee elected to request an exemption from the Technical Specifications pursuant to 10 CFR 72.7 rather than engage in further study to definitively answer the questions raised. By letter dated June 14, 2006, the licensee requested:

  1) an exemption from the wording in the bases section of Technical Specification 1.2.1 that described the transfer cask surface dose rates for the 24P and the 52B canisters. Fort Calhoun was using the 32PT canister.

  2) an exemption from Technical Specification 1.2.11 limiting transfer cask dose rates in its entirety. The transfer cask dose rates could not be met using the bare OS197L transfer cask alone. The use of supplemental shielding, not addressed in the Technical Specification, was required in order to meet the Technical Specification radial dose rate limits.

  3) an exemption from the wording in Technical Specification 1.2.17a concerning start of the vacuum drying time clock. The licensee requested starting the vacuum drying time clock when the first 750 gallons of water was pumped out of the canister, rather than at the initiation of vacuum drying as specified in the Technical Specification. The thermal analysis used for establishing the vacuum drying time limits in Technical Specification 1.2.17a was based on an initial spent fuel assembly cladding temperature of 215 degrees F. The 215 degree F initial cladding temperature was ensured by maintaining a heat sink in the canister of approximately 750 gallons of water until vacuum drying was initiated. The operational sequence that Transnuclear proposed to use at Fort Calhoun would have fully drained the canister 8 to 10 hours prior to vacuum drying, thus eliminating the heat sink and invalidating the 215 degree F initial cladding temperature on which the Technical Specification was based. The generic implications of this issue are currently under review by the Spent Fuel Program Office (SFPO). (Attachment 2, General License)

- The Fort Calhoun Station reactor site parameters were enveloped by the storage cask design parameters contained in the NUHOMS Final Safety Analysis Report (Attachment 2, General License).

- The Fort Calhoun Station 10 CFR 50 and 10 CFR Part 72 conditions were met for dry fuel storage operations (Attachment 2, General License).

Heavy Loads

- All lifts of the transfer cask and canister were made under the Fort Calhoun Station heavy loads requirements and procedures, as documented through a 10 CFR 50.59 evaluation. Slings were used in accordance with ASME Code requirements (Attachment 2, Heavy Loads).

- The lift yoke and transfer cask trunnions were load tested in accordance with the ANSI Standard for special lifting devices (Attachment 2, Lift Yoke and Trunnions).

Quality Assurance

- The licensee's 10 CFR Part 50 Quality Assurance Program had been expanded to envelop the ISFSI. A system of Quality Assurance Program audits, assessments and surveillances had been established and performed to verify compliance with applicable requirements. (Attachment 2, Quality Assurance).

- The licensee had established measures for: a) controlling the calibration of instruments used to verify compliance with the Technical Specifications; b) ensuring conditions adverse to quality were promptly identified and corrected; c) ensuring dry fuel storage components were properly stored to prevent degradation; and d) ensuring purchased material equipment and services conformed to procurement documents (Attachment 2, Quality Assurance).

Radiation Protection

- The licensee had established measures to limit personnel exposures to as low as reasonably achievable (ALARA). Pre-job briefings, temporary shielding, access control measures, low dose waiting areas and dose monitoring were demonstrated during pre-operational testing (Attachment 2, Radiation Protection).

- Criticality control in the canister during loading will be ensured by maintaining a minimum spent fuel pool boron concentration of 2100 ppm. Criticality monitoring systems were installed in all areas where spent fuel was handled (Attachment 2, Radiation Protection).

- The licensee had established a control area around the ISFSI to limit personnel exposures during accident conditions. External dose rate limits for the Horizontal Storage Module were established in accordance with Technical Specifications (Attachment 2, Radiation Protection).

Records

- The licensee had made the required 90 day notification to the NRC prior to loading their first cask, and had established procedural requirements to register the cask with the NRC within 30 days after loading (Attachment 2, Records).

- The licensee had established measures to ensure the 10 CFR 72.212 Report, Certificate of Compliance (and related documents) and the Quality Assurance records were maintained for as long as spent fuel was stored at the ISFSI (Attachment 2, Records).

Safety Evaluations

- The Transnuclear 10 CFR 72.48 evaluation calculated heat removal from the OS197L transfer cask during transport on the transfer trailer with additional shielding. The NRC staff questioned whether the methodology used for the calculation was appropriate for the transfer trailer configuration. Due to time constraints, the licensee elected to request an exemption from this Part 72.48 requirement rather than engage in further study to definitively answer the question raised (Attachment 2, Safety Evaluations).

Training

- The licensee's 10 CFR Part 50 Training Program had been expanded to envelop the ISFSI. All personnel had completed the training and the Fort Calhoun training organization had been assigned maintenance of the ISFSI personnel training records (Attachment 2, Training).

- Pre-operational testing of the ISFSI systems and equipment was completed as required by Technical Specifications. The pre-operational testing comprised the core of the On-The-Job training program for ISFSI personnel (Attachment 2, Training).

Welding and Weld Testing

- Deficiencies in the welding and weld testing procedures that were identified during the pre-operational testing on January 30 through February 2, 2006 had been corrected (Attachment 2, Welding and Weld Testing).

Attachment 1

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee Personnel

S. Andersen - Project Engineer
D. Buell - Fire Protection Program Engineer
G. Cavanaugh - Supervisor, Regulatory Compliance
D. Guinn - Licensing Engineer
R. Haug - Manager, Radiation Protection
L. Hoegen - Radiation Protection Technician
T. Maine - ALARA Coordinator
T. Mathews – Supervisor, Nuclear Licensing
E. Matzke – Licensing Engineer
J. McManis - Manager, Licensing
R. Meng - Senior Emergency Planning Representative
J. Minardi – Working Machinist Leader, Crane Maintenance
R. Paradies - Project Engineer
M. Pohl – Principal Reactor Engineer
R. Ruhge – Supervisor, Quality Control
C. Simmons - Supervisor, Emergency Planning
M. Tesar - Division Manager, Nuclear Support
P. Turner – System Engineer
B. Van Sant - Manager, Nuclear Projects
M. Weeks – Mechanical Engineer
J. Willett - Principal Reactor Engineer
C. Williams - Supervisor, Radiological Operations

TriVis Personnel

D. Bland - Project Manager
J. Kelley – Loading Superintendent
S. Miller - Loading Superintendent
T. Ferrando – Loading Supervisor
L. Wood – Loading Supervisor

Transnuclear Personnel

J. Axline - Project Manager
T. Chen - Quality Assurance Manager
U. Farradj - Project Manager

INSPECTION PROCEDURES USED

60854 Pre-operational Testing of an ISFSI

60856 Review of 10 CFR 72.212(b) Evaluations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Closed

The following deficiencies were identified during the Welding and Fluid Operations demonstration conducted on January 30 through February 2, 2006 and were documented in Inspection Report 050-00285/06-012; 072-00054/06-001 (ML060580267). At the conclusion of this inspection, these deficiencies had been corrected. For details, refer to Attachment 2 of this report.

72-054/0601-01	FIN	Revise the standard temperature liquid penetrant procedure. Develop and qualify a high temperature liquid penetrant procedure (Attachment 2, Welding and Weld Testing).
72-054/0601-02	FIN	Validate the visual testing procedure for both direct and remote testing (Attachment 2, Welding and Weld Testing).
72-054/0601-03	FIN	Develop a method for documenting that the Automated Welding System (AWS) welds are made in accordance with the weld specifications. Calibrate the AWS as specified by the manufacturer (Attachment 2, Welding and Weld Testing).
72-054/0601-04	FIN	Develop a method for performing post testing calibration checks on the vacuum and pressure instruments used for verifying canister dryness and helium back pressure (Attachment 2, Quality Assurance).
72-054/0601-05	FIN	Develop a weld repair procedure that defines the process and provides the required documentation (Attachment 2, Welding and Weld Testing).

Discussed

None.

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
AWS	Automated Welding System
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CMAA	Crane Manufacturers Association of America
EAD	Electronic Alarming Dosimeter
EATL	Energy Absorbing Torque Limiter
FIN	NRC Inspection Finding
FSAR	Final Safety Analysis Report
ft-lbs	Foot pounds
Gwd/MTU	Gigawatt Days per Metric Ton Uranium
GWS	General Welding Standard
HSM	Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
kW	Kilowatt
MCL	Maximum Critical Load
MRS	Monitored Retrievable Storage
NDE	Non-Destructive Examination
NDTT	Nil Ductility Transition Temperature
QA	Quality Assurance
RWP	Radiological Work Permit
SER	Safety Evaluation Report
SSE	Safe Shutdown Earthquake
TEDE	Total Effective Dose Equivalent
WPS	Welding Procedure Specification
wt. %	Weight Percent

Attachment 2

FORT CALHOUN PRE-OPERATIONAL TESTING
Table Of Contents

Attachment 2
FORT CALHOUN PRE-OPERATIONAL TESTING
Inspector Notes

Category: Crane Design Basis **Topic:** Ederer Single-Failure-Proof Criteria

Reference: Ederer Topical Report EDR-1, Section III.C

Requirement The regulatory positions of Regulatory Guide 1.104 have been addressed in the design of Nuclear Safety Related Ederer X-SAM Cranes. Appendices B and C identify the additional plant specific information that is needed to verify a specific retrofit crane's compliance with the regulatory positions.

Finding: This requirement was implemented. During 1981 and 1982 Fort Calhoun replaced the trolley on their auxiliary building crane with an Ederer 75 ton trolley. Fort Calhoun had requested approval from the NRC to designate their retrofitted 75 ton auxiliary building crane as single-failure-proof based on the Ederer Topical Report and the Fort Calhoun plant specific information contained in Appendices B and C. The NRC approved the Fort Calhoun auxiliary building crane as single-failure-proof on March 25, 1981.

Documents Reviewed: NRC Safety Evaluation by the Office of Nuclear Reactor Regulation supporting Amendment No. 57, dated March 25, 1981
Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes", Revision 1

Category: Crane Design Basis **Topic:** NUREG 0612 Phase I & II Letters

Reference: Generic Letters 81-07 and 85-11

Requirement Generic Letter 81-07 required licensees to evaluate their controls for handling heavy loads and to provide these evaluations to the NRC. Generic Letter 85-11 documented that all licensees had submitted a Phase I and a Phase II report, and further stated that while not a requirement, the NRC encouraged the implementation of any actions the licensee identified in Phase II regarding the handling of heavy loads.

Finding: This requirement was implemented. The licensee submitted the requested information to the NRC in two letters dated June 22, 1981 and January 21, 1982. The OPPD letter dated January 21, 1982 included documentation of the single-failure-proof auxiliary building crane in Appendix C. The information was reviewed by the NRC and found to be acceptable, as documented in a letter from the NRC to OPPD dated May 22, 1984.

Documents Reviewed: NRC Generic Letter 81-07 dated December 22, 1980, "Control of Heavy Loads"
OPPD Letters to NRC dated June 22, 1981 and January 21, 1982
NRC Letter to OPPD dated May 22, 1984

Category: Crane Design Basis **Topic:** Seismic Criteria - Load Control

Reference: NUREG 0554, Sect 2.5

Requirement Overhead cranes should be designed to hold and control the load during a seismic event. The bridge and trolley should remain on their respective runways with their wheels prevented from leaving the tracks. Seismically induced pendulum load swing effects on the crane should be considered in the design of the trolley.

Finding: This requirement was implemented. Prior to the use of the auxiliary building crane for dry fuel storage, the licensee contracted with Ederer to verify that the analysis performed in 1980 of crane performance during a Safe Shutdown Earthquake (SSE) was adequate. The analysis found that the crane bridge was always in compression during the SSE and therefore never left the crane rails. However, the analysis was silent on whether the crane rails would leave their support girders. Condition Report #200601545 was generated to resolve the question. Subsequently, Stone and Webster Calculation No. SS-6 confirmed that the crane rail clips were adequate to keep the rails on their support girders during a seismic event.

The Fort Calhoun Updated Safety Analysis Report (USAR) allowed the use of seismic methodology "EA-FC-94-003" for re-analysis of existing structures. This methodology had been reviewed and approved by the NRC. As part of this analysis, Ederer used the seismic methodology "EA-FC-94-003" that provided curves for the various damping values that were used to determine the appropriate value for the crane analysis. The American Society of Mechanical Engineers (ASME) "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge Multiple Girder)," also known as NOG-1-2002, specified that a damping value of 7% be used for the crane analysis when the Safe Shutdown Earthquake (SSE) seismic acceleration values were used. Ederer conservatively used a damping value of 5% when performing the analysis, instead of the 7% value recommended in NOG-1. The analysis evaluated the crane with maximum loads imposed on the crane with the trolley positioned at the end of the span, 1/4 of the span and mid-span. The hook positions analyzed were with the hook up, hook down and hook mid-height. The software program GT STRUDL was used to analyze the data. The analysis concluded that the Fort Calhoun crane structure (bridge and trolley) met the structural requirements of the Crane Manufacturers Association of America (CMAA) Specification #70 and that no structural modification were required for the crane to meet the specified seismic conditions.

Documents Reviewed: Calculation No. FC07190, "Seismic Qualification of the Auxiliary building Crane for Dry Fuel Storage," Revision 0
Stone & Webster Analysis #59058, Revision 1, Calculation No. SS-6
Condition Report #200601545
Fort Calhoun USAR, Appendix F, Section 2.2.3, "Classification of Structures and Equipment Seismic Criteria," Revision 6
Analysis EA-FC-94-003, "Alternate Seismic Criteria and Methodology."

Category: Crane Design Basis **Topic:** Seismic Criteria - Support Structure

Reference: Fort Calhoun Station USAR, Appendix F

Requirement The auxiliary building is designed to Class I seismic criteria. All Class I structures are designed such that the seismic stresses from a maximum hypothetic earthquake, in combination with the primary steady state stresses, will not prevent the safe and orderly shutdown of the plant.

Finding: This requirement was implemented. Two technical reviewers from the Office of Nuclear Reactor Regulation (NRR) reviewed the Stone & Webster seismic analysis for the auxiliary building. The seismic response spectra building between the operating floor at elevation 1025 feet and the building roof at elevation 1082.75 feet, provided evidence that the assumptions used in the mathematical model for the auxiliary building response during a seismic event were adequate. The analysis indicated some overstress conditions in the building columns under a seismic event with a 75 ton load on the crane. However the licensee provided justification that the overstress conditions were the result of over-simplified modeling. The modeling overestimated the mass of the building resulting in an overestimate of the shear force applied to the column. If the model were corrected, the overstress condition would be alleviated. The technical reviewers agreed with the justification and concluded that the auxiliary building was adequate to support a 75 ton crane capacity during a seismic event.

Documents Reviewed: Stone & Webster Analysis #59058, Revision 1
Auxiliary Building construction drawings

Category: Crane Hoist Design **Topic:** Drum Support Structure

Reference: NRC Safety Evaluation Of Ederer Topical Report

Requirement The hoist drum safety support structure consists of a separate hub and stub assembly that is applied to both ends of the drum shell and a restraint structure to prevent the drum gear and emergency brake from disengaging. This assures that the drum will remain in place and hold the load safely in case of a shaft or bearing failure.

Finding: This requirement was implemented. High strength hub assemblies were attached to both ends of the main hoist drum shell. These hub assemblies rested on the safety support structure, which was equipped with a restraint system to prevent the drum gear and emergency brake from disengaging.

Documents Reviewed: Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Revision 1
NRC inspector walkdown of Ederer trolley

Category: Crane Hoist Design **Topic:** Hoist Holding Brakes

Reference: NRC Safety Evaluation Of Ederer Topical Report

Requirement The Ederer hoist incorporates two high speed holding brakes at the hoist motor location. A separate emergency brake is applied directly to the drum, thus eliminating the need for a dual gear train. The emergency drum brake is not used as a holding brake during normal operation.

Finding: This requirement was implemented. The Ederer hoist incorporated two high speed holding brakes. Each brake consisted of a brake drum attached to the gear train which rotated as the hoist was raised and lowered. The brake shoes were stationary and mounted to the crane trolley. When hoist motion stopped, the holding brake shoes engaged the brake drums and the emergency drum brake did not actuate.

Documents Reviewed: Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Revision 1
NRC inspector walkdown of Ederer trolley

Category: Crane Hoist Design **Topic:** Provisions For Manual Operation

Reference: NUREG 0554, Section 3.4

Requirement A crane that has been immobilized because of failure of controls or components while holding a critical load should be able to hold the load or set the load down while repairs or adjustments are made. This can be accomplished by inclusion of features that will permit manual operation of the hoisting system and the bridge and trolley transfer mechanisms by means of appropriate emergency devices.

Finding: This requirement was implemented. Procedure GM-OI-HE-0002, Step 8.9.2 referred to the Ederer Technical Manual for manual operation of the auxiliary building crane. Section 2.6 of the Ederer Technical Manual provided instructions for manually moving the bridge and trolley, lowering the main hoist using the emergency drum brake, and lowering the main hoist using the high speed holding brakes. These manual operations were performed during the pre-operational testing program.

Documents Reviewed: Procedure GM-OI-HE-2, "Auxiliary Building Crane Normal Operation," Revision 15
Ederer Technical Manual TME082.0190, "Refueling Area Crane Mechanical Components," Revision 0
NRC inspector walkdown of Ederer trolley

Category: Crane Hoist Design **Topic:** Wire Rope Breaking Strength

Reference: Ederer Topical Report EDR-1, Appendix B

Requirement The plant specific data provided by Fort Calhoun in EDR-1, Appendix B Supplement stipulated that the auxiliary building crane wire rope would have a diameter of 1 1/4 inches, be 6X37 class IWRC and have a minimum breaking strength of 117,000 pounds.

Finding: This requirement was implemented. The wire rope material certification documented that the wire rope on the auxiliary building crane was 1 1/4 inches in diameter and was classified as 6X37 IWRC with an actual breaking strength of 136,000 pounds.

The Fort Calhoun auxiliary building crane used two independent wire rope systems. In order to meet single-failure-proof criteria, each wire rope system must be capable of holding the full rated static load, plus the dynamic shock loading imposed by a failure of the other wire rope system. The Fort Calhoun auxiliary building crane was rated for a static load of 150,000 pounds. The dynamic shock loading was approximately 2.3 times the static loading, or 345,000 pounds. Therefore each wire rope system must be capable of holding 495,000 pounds.

Each wire rope system on the auxiliary building crane contained 4 sheaves and 8 parts of wire rope. Each vertical length of wire rope is a part. Therefore, the total tension on each wire rope part was 1/8th of the total load of 495,000 pounds, or 61,875 pounds. The minimum wire rope braking strength of 117,000 pounds specified by Ederer was well above the minimum required breaking strength of 61,875 pounds.

Documents Reviewed: Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Appendix B Supplement, Revision 1
Macwhyte Company wire rope certificate dated May 26, 1981

Category: Crane Inspection **Topic:** Annual

Reference: ASME B30.2; Section 2-2.1.3

Requirement Periodic or annual crane inspections shall be performed to check for: a) deformed, cracked or corroded members; b) loose or missing bolts, nuts, pins or rivets; c) cracked or worn sheaves and drums; d) worn, cracked or distorted parts; e) excessive wear of the brake system; f) excessive wear of the drive chain; g) deterioration of controllers or switches; h) inoperable motion limit devices that interrupt power; and i) deteriorated rope reeving system.

Finding: This requirement was implemented. Procedure MM-RI-HE-0551 was used to perform the annual inspection of the auxiliary building crane. The inspection points and acceptance criteria contained in the procedure were consistent with ASME Code B30.2 requirements. The licensee completed the last annual inspection and testing of the auxiliary building crane on February 10, 2006 under Work Order #00221300 and Procedure MM-RI-HE-0551.

Documents Reviewed: Work Order #00221300
Procedure MM-RI-HE-0551, "Annual Inspection of Auxiliary Building Crane," Revision 6

Category: Crane Inspection **Topic:** Frequent

Reference: ASME B30.2; Section 2-2.1.2

Requirement Frequent crane inspections shall be performed to check: a) operating mechanisms; b) upper limit devices; c) air and hydraulic systems; d) hooks and hook latches; e) hoist ropes and end connections; and f) spooling of wire rope onto the drum and sheaves. These inspections should be performed monthly during normal service, weekly to monthly during heavy service and daily to weekly during severe service.

Finding: This requirement was implemented. Procedure GM-OI-HE-0002 was revised under Condition Report #200601495 to incorporate the frequent crane inspections required by ASME Code B30.2. Steps 7.1 through 7.3 were added to require a check of all limit switches for proper operation each shift, a daily check of all the lines, tanks, valves, pumps and other parts of air or hydraulic systems for leakage, and a daily check of the rope reeving system for proper placement of the wire rope in the drum grooves and block sheaves.

Documents Reviewed: Procedure GM-OI-HE-2, "Auxiliary Building Crane Normal Operation," Revision 15
Condition Report #200601495

Category: Crane Inspection **Topic:** Hook

Reference: ASME B30.10, Section 10-1.4

Requirement Hooks shall be inspected for: a) any bend or twist exceeding 10% from the plane of an unbent hook; b) a throat opening greater than 15% of the original opening for hooks without latches or 8% for hooks with latches; c) wear exceeding 10% of the original section dimension; d) cracks, severe nicks or gouges; and e) inoperative latch. These inspections should be performed monthly during normal service, weekly to monthly during heavy service and daily to weekly during severe service.

Finding: This requirement was implemented. Procedure GM-OI-HE-2, Step 8.4.5 required a visual inspection of the main and auxiliary hook prior to the first lift of the day. Evidence of deformation, cracks and damage was cause for rejection. Procedure FCSG-15-25, Step 6.1 required hooks to be inspected prior to each use. Cracks, corrosion, excessive wear in the saddle of the hook, twisting of the hook body by more than 10%, throat opening greater than 15% of original, and an inoperable safety latch were all causes for rejection.

Documents Reviewed: Procedure GM-OI-HE-2, "Auxiliary Building Crane Normal Operation," Revision 15
Procedure FCSG-15-25, "Cranes, Derricks, Rigging and Hoists," Revision 1

Category: Crane Inspection **Topic:** Rope

Reference: ASME B30.2, Sect 2-2.4.1(a)

Requirement All ropes should be visually inspected at the start of each shift. A thorough inspection of all ropes shall be made on a periodic interval (normally annually) that includes the entire length of rope. The inspection certification record shall include the date of inspection, the signature of the person who performed the inspection, and an identifier for the ropes which were inspected.

Finding: This requirement was implemented. Procedure GM-OI-HE-2, Step 7.3 required a visual inspection of the hoist wire rope each day that the crane was used. The licensee performed annual inspections of the wire rope using Procedure MM-RI-HE-0551. The inspection points and acceptance criteria contained in the procedure were consistent with the ASME Code. The licensee completed the last annual inspection of the wire rope on February 10, 2006 under Work Order #00221300. No deficiencies were identified and the inspection certification record was complete.

Documents Reviewed: Procedure GM-OI-HE-2, "Auxiliary Building Crane Normal Operation," Revision 15
Procedure MM-RI-HE-0551, "Annual Inspection Of Auxiliary Building Crane," Revision 6
Work Order #00221300

Category: Crane Load Testing **Topic:** 100% Load Testing

Reference: ASME B30.2, Section 2-2.2.2

Requirement After the 125% static load test, the crane handling system should be given a full performance test with 100% of the Maximum Critical Load (MCL) through all speeds and motions for which the system is designed. This should include verifying all limiting and safety control devices.

Finding: This requirement was implemented. The licensee conducted a 125% rated load test on September 1, 1981. On September 2, 1981 a 100% performance test was performed by traversing the trolley and bridge through all accessible operating positions.

Documents Reviewed: Task FC-76-22, Procedure SRDCO-81-43, "Installation and Testing of the Retro-Fit Trolley System and Overhead Crane Bridge Modifications," dated March 18 and September 27, 1982

Category: Crane Load Testing **Topic:** 125% Load testing

Reference: Ederer Topical Report EDR-1, Section C.1.b(3)

Requirement The crane shall be load tested at 125% of the Maximum Critical Load (MCL) with the ambient temperature at or below the anticipated operating temperature. At completion of the load testing, the existing bridge structure including all accessible welds shall be visually inspected by a competent structural engineer.

Finding: This requirement was implemented. The licensee performed a 125% rated load test on September 1, 1981 and again on March 21, 1982. The test loads were documented to be 125%, +0/-2%, which is within the recommended tolerance of +0/-4% contained in the ASME Code B30.2 interpretations. During the initial load test, the site was unable to achieve a temperature of 50 degrees F or less. The 125% load test was subsequently re-performed on March 21, 1982 with the ambient temperature of the crane at or below 50 degrees F. Procedure GM-OI-HE-2, Steps 6.31 and 7.11 required a minimum temperature of 50 degrees F at the bridge area prior to operating the crane.

During the crane inspection on March 14-15, 2006, the licensee was unable to locate documentation that the visual examination of the crane bridge and its structural welds had been performed following the initial load test. The licensee removed the auxiliary building crane from service on March 16, 2006 and generated Condition Report #200601111 to document the condition. A visual examination of the structural welds was performed under Construction Work Order #05-0066 on March 23, 2006. No indications of cracking or degradation were identified in any of the welds or the base metal structural steel.

Documents Reviewed:
Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Appendix C Supplement, Revision 1
Task FC-76-22, Procedure SRDCO-81-43, "Installation and Testing of the Retro-Fit Trolley System and Overhead Crane Bridge Modifications," dated March 18 and September 27, 1982
Procedure GM-OI-HE-2, "Auxiliary Building Crane Normal Operation," Revision 15
Work Order #1498
Condition Report #200601111

Category: Crane Load Testing **Topic:** Hook Load Testing

Reference: NUREG 0554, Section 4.3

Requirement A static load test should be performed on each load attaching hook at 200% of the Maximum Critical Load (MCL). Measurements of the geometric configuration of the hook should be made before and after the test. Following load testing, volumetric and surface non destructive examinations should be performed to verify the soundness of fabrication and to ensure integrity of the hook.

Finding: This requirement was implemented. On December 9, 1980 a 200% load test was performed on the auxiliary building crane main hook and sister hook. The magnetic particle tests and geometric measurements performed after the load test did not identify any deformation or cracking.

Documents Reviewed: Ederer Hook Pull Test and Magnetic Particle Inspection Reports dated December 9, 1980

Category: Crane Maintenance **Topic:** Corrective Action Program

Reference: 10 CFR 50, Appendix B, XVI

Requirement Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

Finding: This requirement was implemented. The Condition Reports associated with the auxiliary building crane were reviewed dating back to 1995. Two Condition Reports, 200002565 and 200302095, documented malfunctions of the hoist and travel functions. In both cases the crane was repaired and the malfunctions did not recur.

Documents Reviewed: Condition Report No.s 199800373, 200000444, 200002565, 200302095, 200400851, 200402817, and 200600095

Category: Crane Maintenance **Topic:** Preventive Maintenance Program

Reference: ASME B30.2; Section 2-2.3.1

Requirement A preventive maintenance program should be established based on the recommendations outlined in the crane manufacturer's manual.

Finding: This requirement was implemented. Table 3.2.3 of the Ederer Technical Manual required that the upper and lower block sheave bearings, equalizer bearings, hook thrust bearings and flexible couplings be lubricated weekly when in use. Step 7.4 was added to Procedure GM-OI-HE-2 under Condition Report #200601495 to capture these lubrication requirements.

Documents Reviewed: Ederer Technical Manual TME082.0190, "Refueling Area Crane Mechanical Components," Revision 0
Procedure GM-OI-HE-2, "Auxiliary Building Crane Normal Operation," Revision 15
Condition Report #200601495

Category: Crane Safety Systems **Topic:** Energy Absorbing Torque Limiter (EATL)

Reference: Ederer Topical Report EDR-1, Section III.G.3.a

Requirement The Energy Absorbing Torque Limiter (EATL) is a wet type (oil) multi-disc spring loaded clutch which can be adjusted to slip at a prescribed torque. During periodic inspections of the crane, the EATL is tested. The hoist upper travel limit switches are bypassed and the hoist is slowly raised until the EATL clutch slips.

Finding: This requirement was implemented. Testing of the EATL was completed on February 10, 2006 under Work Order #00221300 and Procedure MM-RI-HE-0551. Section 7.16.2 of Procedure MM-RI-HE-0551 adjusted the EATL to slip at the maximum torque that could be developed by the hoist motor, plus the inertia of the rotating parts. The acceptance criteria was 161 to 196 foot-pounds (ft-lbs) and the as-found setting was 165 ft-lbs. No adjustment was necessary.

Section 7.16.5 of Procedure MM-RI-HE-0551 tested the EATL and the drive train continuity detector. With both upper limit switches disconnected, the load block was slowly raised into contact with the load girt until the EATL clutch slipped. When the EATL clutch slipped, the drive train continuity detector sensed a discontinuity between the hoist motor and the drive train, and actuated the Failure Detection System. The Failure Detection System stopped the hoist motor and engaged the emergency drum brake.

Documents Reviewed: Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Revision 1
Work Order #00221300
Procedure MM-RI-HE-0551, "Annual Inspection of Auxiliary Building Crane," Revision 6

Category: Crane Safety Systems **Topic:** Failure Detection System

Reference: Ederer Topical Report EDR-1, Section III.G.3.c

Requirement The Failure Detection System will deenergize the hoist motor and set the emergency brake if one or more sensors detect: a) drive train discontinuity due to EATL actuation; b) actuation of the hoist backup upper limit switch; c) main drum overspeed; or d) wire rope mis-spooling. During periodic inspections of the crane, the Failure Detection System is tested.

Finding: This requirement was implemented. Procedure MM-RI-HE-0551, Section 7.16.4 verified the drive train discontinuity monitor actuated the Failure Detection System on EATL actuation. Failure Detection System actuation on hoist backup upper limit switch actuation was verified during testing of the dead weight upper limit switch in Procedure MM-RI-HE-0551, Section 7.16.7.

Procedure MM-RI-HE-0551, Section 7.16.14 tested the main drum overspeed detector. The overspeed detector shaft was uncoupled from the drum and manually rotated in excess of its setpoint. On detector actuation, power was interrupted to the hoist motor, the emergency drum brake engaged and the overspeed light illuminated on the Failure Detection System panel.

Procedure MM-RI-HE-0551, Section 7.16.15 tested the wire rope spooling monitor. The wire rope spooling bar was manually lifted away from the drum approximately 1 1/4 inches. When the monitor sensed a mis-spooling condition, the emergency drum brake engaged and the rope spooling error light illuminated on the Failure Detection System panel.

Documents Reviewed: Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Revision 1
Work Order #00221300
Procedure MM-RI-HE-0551, "Annual Inspection of Auxiliary Building Crane," Revision 6

Category: Crane Safety Systems **Topic:** Hoist Emergency Drum Brake

Reference: Ederer Topical Report EDR-1, Section III.G.3.a

Requirement During periodic inspections of the crane, all automatic and manual actuations of the emergency drum brake are tested. The emergency drum brake is automatically actuated by the Failure Detection System. The emergency drum brake is manually actuated by depressing the emergency stop pushbutton.

Finding: This requirement was implemented. All actuations of the emergency drum brake were tested on February 10, 2006 under Work Order #00221300 and Procedure MM-RI-HE-0551. Procedure MM-RI-HE-0551, Section 7.16.14 verified the overspeed detector actuated the Failure Detection System on drum overspeed. Section 7.16.4 verified the drive train continuity monitor actuated the Failure Detection System on EATL actuation. Section 7.16.15 verified the wire rope spooling monitor actuated the Failure Detection System on wire rope mis-spooling. Section 7.16.7 of Procedure MM-RI-HE-0551 verified the hoist backup upper limit switch actuated the Failure Detection System when tripped.

Manual testing of the emergency drum brake was performed under Section 7.16.6 of Procedure MM-RI-HE-0551. At the test panel on top of the trolley, the emergency drum brake control was placed in MANUAL. The brake was opened with air pressure and power to the crane was verified to be uninterrupted. The brake was then closed from the test panel and power to the crane was verified to be interrupted. Emergency drum brake control was then returned to AUTOMATIC.

Documents Reviewed: Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Revision 1
Work Order #00221300
Procedure MM-RI-HE-0551, "Annual Inspection of Auxiliary Building Crane," Revision 6

Category: Crane Safety Systems **Topic:** Hoist Upper Travel Limit Switches

Reference: Ederer Topical Report EDR-1, Section III.G.1.a

Requirement The Ederer hoist incorporated two separate upper travel limit switches; a rotary switch and a dead weight switch. On upward hoist motion, the rotary switch (primary) is the first to actuate followed by the dead weight switch (backup). During periodic inspections of the crane, both upper travel limit switches are tested.

Finding: This requirement was implemented. Testing of the upper travel limit switches was completed on February 10, 2006 under Work Order #00221300 and Procedure MM-RI-HE-0551. Procedure MM-RI-HE-0551, Section 7.16.7 tested the dead weight switch. With the rotary switch (primary) disconnected, the hoist was raised until the dead weight switch actuated to stop hoist upward motion. The Failure Detection System was reset, the load block was lowered, and the rotary switch was re-connected.

Procedure MM-RI-HE-0551, Section 7.16.8 tested the rotary switch. With the dead weight switch (backup) connected, the hoist was raised until the rotary switch actuated to stop hoist upward motion.

Documents Reviewed: Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Revision 1
Work Order #00221300
Procedure MM-RI-HE-0551, "Annual Inspection of Auxiliary Building Crane," Revision 6

Category: Crane Safety Systems **Topic:** Hydraulic Load Equalization System

Reference: Ederer Topical Report EDR-1, Section III.G.2.b

Requirement The Ederer crane incorporates a hydraulic load equalization system that balances the load between the two wire rope systems and cushions the shock to either wire rope system on a failure of the other. The load equalization system is tested and sealed at the manufacturer. During periodic inspections the hydraulic fluid level is checked by monitoring the oil pressure gauge.

Finding: This requirement was implemented. The licensee completed the last annual inspection of the hydraulic load equalization system on February 10, 2006 under Work Order #00221300 and Procedure MM-RI-HE-0551. The inspection identified that the hydraulic system fluid level was low. The as-found system pressure was 15 psig rather than the required 50 to 250 psig. The system leaks were repaired and the hydraulic fluid level and pressure were restored under Work Request #90347.

Documents Reviewed: Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Revision 1
Work Order #00221300
Procedure MM-RI-HE-0551, "Annual Inspection of Auxiliary Building Crane," Revision 6
Work Request #90347

Category: Crane Safety Systems **Topic:** Overload Protection

Reference: Ederer Topical Report EDR-1, Section III.G.1.c

Requirement The hoist motor is protected from overload by a load cell installed in the hoist reeving system. During periodic inspections of the crane, the load cell and overload protection system is tested.

Finding: This requirement was implemented. Testing of the hoist overload system was completed on February 10, 2006 under Work Order #00221300 and Procedure MM-RI-HE-0551, Section 7.16.1. Both the rotary and dead weight upper limit switches were disconnected. The load block was raised slowly until it contacted the bottom of the load girt. Both holding brakes were engaged and power was removed from the hoist motor. The outboard holding brake was clamped open and a torque wrench was attached to the brake wheel nut. The torque wrench was then used to raise the tension on the wire rope to the required overload setpoint of 136 ft-lbs. When power was restored to the hoist motor, it was immediately interrupted by the overload protection system.

Documents Reviewed: Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Revision 1
Work Order #00221300
Procedure MM-RI-HE-0551, "Annual Inspection of Auxiliary Building Crane," Revision 6

Category: Crane Safety Systems **Topic:** Travel Interlocks Near The Spent Fuel Pool

Reference: Fort Calhoun Station USAR, Section 14.24.1.2

Requirement The auxiliary building crane is provided with an electrical interlock system that will normally prevent the trolley from moving over the spent fuel pool. The interlocks may be bypassed provided a crane supervisor is present to direct crane operation.

Finding: This requirement was implemented. Electrical interlocks were provided to stop auxiliary building crane travel when the main hoist approached the spent fuel pool boundary on the north, south and west sides. Access from the east was prevented by the south boundary interlock. Procedure MM-RI-HE-0551, Step 7.8 tested operability of these bridge and trolley interlocks annually. A key operated bypass switch was provided in the crane cab and on the remote control box for use during fuel loading operations. The key was controlled by the on-duty shift manager.

Documents Reviewed: Procedure MM-RI-HE-0551, "Annual Inspection of the Auxiliary Building Crane," Revision 6
NRC inspector walkdown of Ederer trolley

Category: Emergency Planning **Topic:** Emergency Plan

Reference: 10 CFR 72.32(c)

Requirement For an ISFSI that is located on the site of a nuclear power plant licensed for operation, the Emergency Plan required by 10 CFR 50.47 shall be deemed to satisfy the requirements of this section.

Finding: This requirement was implemented. The ISFSI was located on the reactor site and had been included in the licensee's 10 CFR 50.47 Emergency Plan. The licensee added Emergency Action Level (EAL) 7.1, "Damage to a Loaded Cask Confinement Boundary," which was classified as a Notification of Unusual Event. Other existing EALs adequately addressed Emergency Plan requirements for fire, explosion, flooding, tornado and earthquake events.

Documents Reviewed: Radiological Emergency Response Plan, Revision 33

Category: Emergency Planning **Topic:** Emergency Plan Training

Reference: 10 CFR 50, Appendix E, Section F.1

Requirement The emergency program shall provide for the training of employees and exercising, by periodic drills, of radiation emergency plans to ensure that employees are familiar with their specific emergency response duties.

Finding: This requirement was implemented. The licensee had issued a "Read and Acknowledge" form, along with ISFSI related Emergency Plan changes, to all of the site responders. A forthcoming drill was planned to include the ISFSI.

Documents Reviewed: Hotline Training, HL06-400

Category: Fire Protection **Topic:** Control of Combustible Materials

Reference: FSAR 1004, Section M.4.6.3

Requirement The postulated worst case fire accident is a 300 gallon diesel fuel fire engulfing the transfer cask for 15 minutes at a temperature of 1,475 degrees F. Combustible materials in proximity to a loaded transfer cask should be controlled such that a fire involving all of the combustible materials will not exceed the bounding fire conditions.

Finding: This requirement was implemented. Procedure RE-RR-DFS-0003, Step 5.11 and

Attachment 9.1 controlled fire and explosion sources below the amount needed for the bounding fire. Prior to moving the transfer trailer to the ISFSI, a complete walkdown of the haul path was conducted in accordance with Attachment 9.1 of Procedure RE-RR-DFS-0003. One oxygen/acetylene bottle was found uncapped in the weld shop and was capped. Two vehicles were identified within 30 feet of the haul path and were removed. Two gasoline sources (pressure washer and ground tamper) were identified within 50 feet of the haul path and were removed. A qualified fire watch, with a two fire extinguisher cart, walked alongside the prime mover during transfer from the auxiliary building to the ISFSI. A walkdown of the ISFSI pad identified several large pallets of radwaste stored on the east end of pad. The licensee stated that the radwaste would be removed from the ISFSI pad prior to first loading.

Documents Reviewed: Procedure RE-RR-DFS-0003, "Loaded DSC/TC From Auxiliary Building to ISFSI Operations," Revision 1

Category: Fire Protection **Topic:** External Explosion

Reference: FSAR 1004, Section 3.3.6

Requirement Externally initiated explosions are considered to be bounded by the design basis tornado generated missile load analysis. Licensees are required by 10 CFR Part 72, Subpart K to confirm that no conditions exist near the ISFSI that would result in pressures from an explosion exceeding those postulated for tornado missile or wind effects.

Finding: This requirement was implemented. Section 4.0 of the Fire Hazards Analysis Manual evaluated the potential explosion hazards at the ISFSI and along the haul route between the ISFSI and the Auxiliary Building. These explosion hazards were credible but were bounded by the tornado missile and wind effects presented in Section 8.2.2 of the NUHOMS FSAR. The NUHOMS system was evaluated for a tornado wind velocity of 360 mph with a translational velocity of 70 mph and a pressure drop of 3 psi.

Hydrogen cylinders were stored on the west side of warehouse #13. These cylinders were located 152 feet from the nearest Horizontal Storage Module (HSM) and 102 feet from the south approach slab to the ISFSI. These distances precluded blast pressures from exceeding the design basis tornado forces at the ISFSI or along the haul route. Propane cylinders were stored on the south side of the new warehouse and at the propane cylinder farm on the north side of the maintenance shop. Both the ISFSI pad and the haul route were shielded from the blast pressures from these gas cylinders by the new concrete warehouse.

The Bechtel warehouse and maintenance shed contained oxygen and acetylene gas cylinders used for welding. These gas cylinders were located 330 feet from the nearest HSM, however the haul route carried the transfer cask to within 50' of the gas cylinders during transfer operations. In order to eliminate this explosion hazard, Procedure RE-RR-DFS-0003, Attachment 9.1, Step 5.0 prohibited welding operations in either the Bechtel warehouse or maintenance shed during transfer operations. Further, all gas cylinders were required to be restrained with their outlet valves shut and protective caps in place.

Documents Reviewed: Fire Hazards Analysis Manual EA-FC-97-001, Appendix D - "Independent Spent Fuel

Storage Installation Fire and Explosion Hazards Review," Revision 9
Procedure RE-RR-DFS-0003, "Loaded DSC From Auxiliary Building to ISFSI Operations," Revision 1

Category: Fire Protection **Topic:** Fire Protection Plan

Reference: 10 CFR 50.48(a)(1)

Requirement Each operating nuclear power plant must have a fire protection plan that satisfies Criterion 3 of Appendix A to Part 50. This fire protection plan must describe the overall fire protection program for the facility.

Finding: This requirement was implemented. The Fort Calhoun Station Fire Protection Plan and Station Fire Plan were both revised to incorporate the ISFSI. Training requirements for the station fire brigade were identified and training was provided.

Documents Reviewed: Procedure SO-G-102, "Fire Protection Plan," Revision 7
Procedure SO-G-28, "Station Fire Plan," Revision 63

Category: Fire Protection **Topic:** Offsite Emergency Support

Reference: 10 CFR 72.122(g)

Requirement Structures systems and components important to safety must be designed for emergencies. The design must provide accessibility to emergency equipment, facilities and services such as hospitals, fire and police departments, ambulance services, and other emergency agencies.

Finding: This requirement was implemented. The ISFSI design provided direct access to emergency equipment, facilities and services, including ambulance and fire response vehicles. The licensee had met with the local fire departments and had briefed them on the inclusion of the ISFSI into the Fort Calhoun Fire Protection plan. Fort Calhoun planned to include additional tours and information during future meetings with the local fire departments.

Documents Reviewed: Procedure SO-G-102, "Fire Protection Plan," Revision 7
Procedure SO-G-28, "Station Fire Plan," Revision 63

Category: Fuel Selection/Verification **Topic:** Allowable Fuel For Storage

Reference: CoC 1004, Tech Spec 1.2.1

Requirement The characteristics of the spent fuel allowed to be stored in the NUHOMS 32PT system are limited by Tables 1-1e, 1-1f, 1-1g, and 1-2f.

Finding: This requirement was implemented. The licensee's planned loading campaign consisted of ten canisters containing 32 spent fuel assemblies each. The licensee elected to load only intact Combustion Engineering (CE) 14X14 fuel assemblies with zircalloy cladding and no Poison Rod Assemblies or Burnable Poison Rod Assemblies.

Technical Specification Table 1-1e contained the specifications for fuel assemblies authorized for storage in the 32PT canister. Only intact fuel assemblies with zircalloy cladding were authorized. The intact determination was made using Procedure

RE-AD-0004. Spent fuel assemblies discharged from an operating cycle with reactor coolant chemistry records indicating no fuel failures, were classified as intact. Spent fuel assemblies from an operating cycle with reactor coolant chemistry records indicating possible fuel failures were classified as "suspect", pending supplemental inspection (sipping, ultrasonic testing or eddy current testing). Based on the supplemental inspection results, these "suspect" fuel assemblies were reclassified as either intact or damaged. Visual inspections were performed to identify structural damage. Intact fuel assemblies were limited to structural damage that would not preclude fuel assembly handling by normal means. The intact classification for each fuel assembly was documented in Attachment 8.2 of Procedure RE-AD-0004 and in Attachment 3 of Procedure RE-ST-DFS-0001.

Technical Specification Table 1-1f limited each fuel assembly to a maximum unirradiated length of 165.75 inches, a maximum loading of 0.475 metric tons of uranium dioxide, a maximum of 176 fuel rods and a maximum of 5 guide/instrument tubes. Each fuel assembly selected for loading met the requirements of Table 1-1f, as documented in Attachment 3 of Procedure RE-ST-DFS-0001.

Technical Specification Table 1-1g contained the initial enrichment specifications. The licensee had procured nine NUHOMS Type A canisters with the 16 poison plate configuration and one Type B canister with the 24 poison plate configuration. Table 1-1g specified a maximum initial enrichment of 3.90 wt.% for fuel assemblies stored in the Type A canisters and a maximum initial enrichment of 4.70 wt.% for fuel assemblies stored in the Type B canister. The initial enrichment values for the fuel assemblies selected for loading ranged between 1.379 wt.% and 3.599 wt.%, as documented in Step 6.4 and in Attachment 3 of Procedure RE-ST-DFS-0001.

Technical Specification Table 1-2f limited fuel assembly burnup to 45 gigawatt days per metric ton of uranium (Gwd/MTU). The corrected burnup values for the fuel assemblies selected for loading ranged between 7.398 GWd/MTU and 39.902 Gwd/MTU, as documented in Attachments 2 and 3 of Procedure RE-ST-DFS-0001.

Technical Specification Table 1-2f established minimum cooling times for fuel assemblies as a function of burnup, initial enrichment and decay heat values. The fuel assemblies selected for the loading campaign had burnup values between 7.398 and 39.902 Gwd/MTU, maximum initial enrichment values between 1.379 and 3.599 wt.%, and decay heat values between of 0.092 and 0.687 kW. For these fuel assemblies, Table 1-2f established minimum cooling times between 5 and 11 years. The actual cooling time for each fuel assembly was calculated and verified to meet its minimum cooling time, as documented in Attachment 3 of Procedure RE-ST-DFS-0001.

The NRC exemption will limit the licensee's actual loading campaign to four canisters containing spent fuel assemblies with lower decay heat values and greater cooling times. These spent fuel assemblies will be verified to meet Technical Specifications and the NRC exemption requirements prior to loading.

Documents Reviewed: Procedure RE-AD-0004, "Fuel Characterization of Spent Fuel For Dry Storage," Revision 1

Procedure RE-ST-DFS-0001, "Fuel Selection Verification For Placement in Dry Fuel Storage," Revision 0

Category: Fuel Selection/Verification **Topic:** Cask Loading Plan

Reference: FSAR 1004, Section M.8.1.2.5

Requirement A cask loading plan shall be developed to verify the fuel assemblies meet the burnup, enrichment, and cooling time parameters of Technical Specification 1.2.1. The loading plan shall be independently verified and approved before the fuel load. A fuel movement schedule shall be written, verified and approved based on the loading plan. All fuel movements from any rack location shall be performed under strict compliance with the fuel movement schedule.

Finding: This requirement was implemented. Technical Specification Table 1-1e referred to Figures 1-2, 1-3, and 1-4 for acceptable canister loading configurations. The licensee elected to use Heat Load Zoning Configuration 3 (Technical Specification Figure 1-4), which limited all 32 canister cells to a maximum heat load of 0.7 kW per cell. The licensee elected to further restrict the 20 peripheral cells of the canister to a maximum heat load of 0.5 kW per cell for site dose considerations. Procedure RE-AD-0005, Attachment 8.3 contained an illustration of the acceptable loading configuration.

Procedure RE-AD-0005 was used to develop the canister loading plan. The 32 fuel assemblies selected for loading into each canister, and their assigned canister cell locations, were documented in Attachment 8.1 of Procedure RE-AD-0005. Each fuel assembly decay heat load was verified to be within the limit for each cell, as specified in Attachment 8.3. The total heat load for each canister was then calculated and verified to be less than 18.4 kW.

The minimum spent fuel pool soluble boron concentration required for loading was based on the maximum enrichment of the fuel to be loaded, in conjunction with the canister poison plate configuration. Technical Specification Table 1-1e referred to Table 1-1g for spent fuel pool soluble boron concentration requirements. The licensee had procured nine NUHOMS Type A canisters with the 16 poison plate configuration and one Type B canister with the 24 poison plate configuration. The highest fuel assembly enrichment in the planned loading campaign was 3.599 wt.%. For this enrichment and canister poison plate combination, Table 1-1g specified a minimum spent fuel pool boron concentration of 2100 ppm. The loading plan and minimum spent fuel pool boron concentration requirements were independently reviewed and approved by the Principal Reactor Engineer, as documented in Attachment 8.1 of Procedure RE-AD-0005.

Procedure NMA-3, Step 4.2.4 required that all spent fuel movements between the spent fuel pool racks and the dry fuel storage canister be made by Operations personnel or designated fuel handlers using Form F-2, "Fuel Handling Checklist." The Fuel Handling Checklist for each canister provided the sequence for moving each fuel assembly, its "from" and "to" locations, and its required orientation in the canister. Each checklist was reviewed and approved by the Principal Reactor Engineer, as documented on Form F-1.

Documents Reviewed: Procedure RE-AD-0005, "Fuel Selection and DSC Planning For Dry Cask Storage," Revision 0
Procedure NMA-3, "Special Nuclear Material Control and Accountability," Revision 13
Form F-1,"Fuel Handling Checklist Coversheet," Revision 0
Form F-2, "Fuel Handling Checklist," Revision 4

Category: General License **Topic:** Certificate of Compliance Conditions

Reference: 10 CFR 72.212(b)(2)(i)(A)

Requirement A general licensee shall perform written evaluations, prior to use, that establish that the conditions set forth in the Certificate of Compliance have been met.

Finding: This requirement was not fully implemented. During onsite inspections at the Fort Calhoun Station the NRC staff identified concerns with the proposed use of the Transnuclear lightweight transfer cask designated OS197L. The OS197L transfer cask had been determined to be acceptable for use by the licensee using the 10 CFR 72.212 process. The NRC-identified concerns led OPPD to the conclusion that an exemption was needed for use of this transfer cask at the Fort Calhoun Station. The Fort Calhoun CoC 1004 Exemption Request was received by the NRC on June 9, 2006 and is under review. The exemption request included a number of issues related to Technical Specification compliance. These included:

1) an exemption from the wording in the bases section of Technical Specification 1.2.1 that described the transfer cask surface dose rates for the 24P and the 52B canisters. The licensee had selected the 32PT canister for dry fuel storage at the Fort Calhoun Station.

2) an exemption from Technical Specification 1.2.11 in its entirety. The transfer cask dose rates could not be met using the bare OS197L transfer cask alone. The use of supplemental shielding, not addressed in the Technical Specification, was required in order to meet the Technical Specification radial dose rate limits.

3) an exemption from the wording in Technical Specification 1.2.17a that started the vacuum drying time clock at the initiation of vacuum drying. The licensee requested starting the vacuum drying time clock when the first 750 gallons of water was pumped out of the canister in the spent fuel pool, rather than at the initiation of vacuum drying as specified in the Technical Specification. The thermal analysis used for establishing the vacuum drying time limits in Technical Specification 1.2.17a was based on an initial spent fuel cladding temperature of 215 degrees F. The 215 degree F initial clad temperature was ensured by maintaining a heat sink in the canister of approximately 750 gallons until vacuum drying was initiated. The operational sequence that Transnuclear proposed to use at Fort Calhoun would have fully drained the canister 8 to 10 hours prior to vacuum drying, thus invalidating the 215 degree F initial clad temperature on which the Technical Specification was based. The generic implications of this issue are currently under review by the Spent Fuel Program Office (SFPO).

If the exemptions are approved, the 72.212 Report and the loading procedures will be revised by the licensee to incorporate the provisions of the exemptions.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.
Request For Exemption From NUHOMS Certificate of Compliance No. 1004, Amendment No. 8 transmitted to the NRC under OPPD letter LIC-06-0056 dated June 9, 2006

Category: General License **Topic:** FSAR Conditions

Reference: 10 CFR 72.212(b)(3)

Requirement The general licensee shall review the FSAR referenced in the CoC and the related NRC Safety Evaluation Report, prior to use of the general license, to determine whether or not the reactor site parameters, including analysis of earthquake intensity and tornado missiles, are enveloped by the cask design basis considered in these reports. The results of this review must be documented in the evaluation made in 10 CFR 72.212(b)(2).

Finding: This requirement was implemented. Section 8.0 of the 10 CFR 72.212 Report listed all of the reactor site parameters together with an evaluation of how they were bounded by the cask design. The reactor site parameters for temperature, seismic acceleration, flood, fire and explosion, lightning, tornado, and tornado generated missiles were enveloped by the NUHOMS system design parameters. The site design basis tornado wind speed was verified to be 300 mph, which was bounded by the NUHOMS system design of 360 mph.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.

Category: General License **Topic:** FSAR Conditions - Flood

Reference: CoC 1004, Tech Spec 1.1.1.4

Requirement The site specific analyzed flood condition shall be no greater than 15 feet per second water velocity and a height of 50 feet of water (full submergence of the loaded HSM). This evaluation may be included in the 72.212(b) evaluation report.

Finding: This requirement was implemented. Section 8.2.2 of the 10 CFR 72.212 Report documented that the probable maximum flood occurred at an elevation of 1009.3 feet, as described in the Fort Calhoun Station Updated Safety Analysis Report (USAR). The ISFSI base mat elevation was 1009.83 feet and therefore not subject to flooding.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.

Category: General License **Topic:** FSAR Conditions - HSM Placement

Reference: CoC 1004, Tech Spec 1.1.10

Requirement A minimum of two (2) HSM-Hs are required to be placed adjacent to each other for stability during design basis flood loads.

Finding: This requirement was implemented. The ten Horizontal Storage Modules were placed on the ISFSI pad in sets of two, placed back-to-back. Each set of two HSMs was placed adjacent to the next, creating a 2 X 5 block of HSMs.

Documents Reviewed: None.

Category: General License **Topic:** FSAR Conditions - Lightning Damage

Reference: CoC 1004, Tech Spec 1.1.1.7

Requirement The potential for lightning damage to any electrical system associated with the standardized NUHOMS system should be addressed based on site specific considerations. This evaluation may be included in the 72.212(b) evaluation report.

Finding: This requirement was implemented. Section 8.2.5 of the 10 CFR 72.212 Report addressed the potential for lightning damage to electrical systems associated with the NUHOMS system. At the Fort Calhoun Station ISFSI, the only electrical system that could be adversely affected by lightning was the temperature monitoring system. However, if this system were damaged, the NUHOMS Technical Specification allowed for an alternate method of ensuring proper thermal performance of the HSMs.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.

Category: General License **Topic:** FSAR Conditions - Seismic Acceleration

Reference: CoC 1004, Tech Spec 1.1.1.3

Requirement The site specific horizontal seismic acceleration level shall be 0.25g or less. The site specific vertical seismic acceleration level shall be 0.17g or less. This evaluation may be included in the 72.212(b) evaluation report.

Finding: This requirement was implemented. Section 8.2.4 of the 10 CFR 72.212 Report documented that the site earthquake acceleration levels were bounded by Technical Specification 1.1.1.3.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.

Category: General License **Topic:** FSAR Conditions - Site Average Temperatures

Reference: CoC 1004, Tech Spec 1.1.1.1

Requirement The site average yearly temperature with solar incidence shall be 70 degrees F or less. The site average daily temperature shall be 100 degrees F or less. This evaluation may be included in the 72.212(b) evaluation report.

Finding: This requirement was implemented. Section 8.2.1.1 of the 10 CFR 72.212 Report documented that the mean annual temperature for the Blair, Nebraska region was 51.1 degrees F. This information was derived from Section 2.5.1 of the Fort Calhoun USAR.

Section 8.2.1.2 of the 10 CFR 72.212 Report documented that for the time period 1961 through 1990, the maximum daily temperature was 87.7 degrees F. This information was derived from Table 2.5-5 of the Fort Calhoun USAR.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.

Category: General License **Topic:** FSAR Conditions - Site Temperature Extremes

Reference: CoC 1004, Tech Spec 1.1.1.2

Requirement For Horizontal Storage Modules containing 32PT canisters, the site specific temperature extremes shall be minus 40 degrees F with no solar incidence and plus 117 degrees F with solar incidence. The 117 degree F temperature corresponds to a 24-hour calculated average temperature of 102 degrees F. This evaluation may be included in the 72.212(b) evaluation report.

Finding: This requirement was implemented. Section 8.2.1.3 of the 10 CFR 72.212 Report documented that the Omaha area experienced a record high of 114 degrees F in July 1936 and a record low of minus 32 degrees F in January 1885. For an HSM containing a 32PT canister these extremes are bounded by an upper limit of 117 degrees F and a lower limit of minus 40 degrees F.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.

Category: General License **Topic:** Part 50 Conditions

Reference: 10 CFR 72.212(b)(4)

Requirement Prior to use of the general license, determine whether activities related to storage of spent fuel involve a change in the facility technical specifications or require a license amendment for the facility pursuant to Part 50.59(c)(2). Results of this determination must be documented in the evaluation made in 10 CFR 72.212(b)(2).

Finding: This requirement was implemented. Section 9.0 of the 10 CFR 72.212 Report documented that a review of 10 CFR 50.59 evaluations was conducted to determine if the ISFSI would require a change to the facility Technical Specifications or a license amendment. Table 9.1 provided a listing of ten evaluations that were performed. Evaluation EC-37848 resulted in the generation of License Amendment Request 05-013, "Criticality Control During Spent Fuel Cask Loading in the Spent Fuel Pool".

License Amendment #239 was approved by the NRC on April 10, 2006 and the NRC Safety Evaluation Report required a minimum spent fuel pool boron concentration of 800 parts per million (ppm) to maintain subcriticality during cask loading operations. Procedure RE-RR-DFS-0001, Steps 7.4.1 through 7.4.3, directed chemistry to collect two samples of spent fuel pool water and analyze them for boron concentration. The loading supervisor and the reactor engineer or shift manager were required to validate that the sample results were greater than those specified in Table 1-1g of the Technical Specifications. Table 1-1g specified a minimum spent fuel pool boron concentration of 2100 ppm for the fuel assemblies selected for the planned loading campaign.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.
License Amendment #239, "Criticality Control During Spent Fuel Cask Loading and Unloading Operations in the Spent Fuel Pool," dated April 10, 2006 (ML061000606)
Procedure RE-RR-DFS-0001, "DSC/TC Prep for Fuel Loading Operations," Revision 0

Category: General License **Topic:** Part 50 Conditions - Program Effectiveness

Reference: 10 CFR 72.212(b)(6)

Requirement The general licensee shall review the reactor emergency plan, quality assurance program, training program and radiation protection program to determine if their effectiveness is decreased and if so, prepare the necessary changes and seek and obtain the necessary approvals.

Finding: This requirement was implemented. Section 11.2.1 of the 10 CFR 72.212 Report evaluated the reactor emergency plan; Section 11.2.2 evaluated the quality assurance program; Section 11.2.3 evaluated the training program; and Section 11.2.4 evaluated the Radiation Protection Plan. The licensee concluded that the effectiveness of these programs was not reduced or decreased as a result of activities associated with ISFSI operations.

Documents Reviewed: 10 CFR 72.212, Report, Revision Prelim.

Category: General License **Topic:** Part 72 Conditions - Effluents & Direct Radiation

Reference: 10 CFR 72.212(b)(2)(i)(C) & 10 CFR 72.104(a)

Requirement The general licensee shall perform a written evaluation that establishes that the requirements of 10 CFR 72.104, "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI", have been met. 10 CFR 72.104 requires the annual dose equivalent to any real individual located beyond the controlled area must not exceed 25 mrem to the whole body during normal operations and anticipated occurrences.

Finding: This requirement was implemented. The licensee had calculated the annual offsite dose to an individual located 137.2 meters from the ISFSI for 8760 hours per year (continuous occupancy). The nearest distance from the ISFSI to the Missouri River, where a member of the public could be present, was approximately 480 feet or 146.3 meters. Therefore, the dose calculation was conservative. The calculated dose was 6.1 millirem whole body due to direct radiation, 0.18 millirem to the the thyroid and 1.83 to other critical organs. The doses to the thyroid and other critical organs were calculated from the direct radiation dose by using the organ dose weighting factors. When the contributions from the operating plant were included, the whole body dose was 11.6 millirem, thyroid 6.5 millirem, and other critical organs 9.6 millirem. These values were below the 10 CFR 72.104 limits of 25, 75 and 25 millirem respectively.

There were no effluents emanating from ISFSI operations. The licensee had established a direct radiation monitoring program around the ISFSI. Four thermoluminescent dosimeters had been placed on the four cardinal directions on, or adjacent to, the fences surrounding the ISFSI. These four dosimeters were in addition to the existing Fort Calhoun Station environmental monitoring dosimeters and were being exchanged on a six month interval.

Documents Reviewed: Calculation 1121-0502, "OPPD ISFSI Phase I Site Dose and Occupational Dose Summary," Revision 1
Drawing No. 59058-EY-1A-0, "Independent Spent Fuel Storage Installation Site Location Plan"

Category: General License **Topic:** Part 72 Conditions - First System In Service

Reference: CoC 1004, Tech Spec 1.1.7

Requirement The heat transfer characteristics of the first Horizontal Storage Module (HSM) placed in service, and of each subsequent HSM with higher decay heat load, will be determined by temperature measurements taken at the air inlet and outlet. The heat transfer characteristics of HSMs with lower decay heat loads need not be determined. A letter report summarizing the results of the measurements shall be submitted to the NRC within 30 days of placing the HSM in service.

Finding: This requirement was implemented. Attachment A of the 10 CFR 72.212 Report stated that the first NUHOMS HSM-H system was placed in service at Progress Energy's H.B. Robinson Plant in 2005. The letter summarizing the heat transfer characteristics was submitted to the NRC on September 9, 2005. The canisters planned for loading at Fort Calhoun contained decay heat values lower than the canisters loaded at the H.B. Robinson Plant.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.

Category: General License **Topic:** Part 72 Conditions - Supplemental Shielding

Reference: CoC 1004, Tech Spec 1.1.9

Requirement Supplemental shielding and engineered features (e.g., earthen berms, shield walls) that are used to ensure compliance with 10 CFR 72.104(a) are considered to be important to safety and must be appropriately evaluated under 10 CFR 72.212(b).

Finding: This requirement was implemented. Attachment A of the 10 CFR 72.212 Report stated that no additional engineering features beyond those included in the Transnuclear licensed design were required for Fort Calhoun Station to meet the 10 CFR 72.104(a) requirements.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.

Category: General License **Topic:** Part 72 Conditions - Surveillance Frequencies

Reference: CoC 1004, Tech Spec 1.1.8

Requirement The specified frequency for each Surveillance Requirement is met if the surveillance is performed within 1.25 times the interval specified, as measured from the previous performance.

Finding: This requirement was implemented. Procedure SO-G-23, Section 5.3.12 applied the 10 CFR 50 surveillance requirements to dry fuel storage. Step 5.3.12.B identified the 1.25 times interval in a manner consistent with Technical Specification 1.1.8.

Documents Reviewed: Procedure SO-G-23, "Surveillance Test Program," Revision 53

Category: Heavy Loads **Topic:** Heavy Loads Safety Review

Reference: CoC 1004, Tech Spec 1.1.4

Requirement Lifts of the canister and transfer cask must be made within the existing heavy loads requirements and procedures of the licensed nuclear plant. A safety review under 10 CFR 50.59 is required to show operational compliance with NUREG 0612 and/or the existing plant specific heavy loads requirements.

Finding: This requirement was implemented. Section III of EC No. 32306 stated that dry cask loading operations will be conducted in accordance with the Fort Calhoun Station heavy loads program. Load drops due to crane failure, rigging failure or human error were prevented through use of a single-failure-proof crane, identification of safe load paths, use of load handling procedures, periodic inspection and testing of the crane, qualification and training of the operators, and use of special lifting devices qualified in accordance with ANSI Standard N14.6.

Documents Reviewed: 50.59 Screen - RAMS EC No. 32306, "FCS ISFSI Activities," Revision 2, dated March 24, 2006

Category: Heavy Loads **Topic:** Maximum Lift Height

Reference: CoC 1004, Tech Spec 1.2.13

Requirement When canister basket temperature is below minus 20 degrees F, the transfer cask shall not be lifted inside the spent fuel building. When canister basket temperature is between minus 20 degrees F and 0 degrees F, the maximum transfer cask lifting height is 80 inches. When canister basket temperature is greater than 0 degrees F, no lifting height limits are imposed.

Finding: This requirement was implemented. Procedure RE-RR-DFS-0003, Step 5.14.2 provided the transfer cask lifting restrictions imposed at low temperatures. The restrictions were consistent with Technical Specification 1.2.13.

Documents Reviewed: Procedure RE-RR-DFS-0003, "Loaded DSC/TC From Auxiliary Building to ISFSI Operations," Revision 1

Category: Heavy Loads **Topic:** Minimum Lift Height

Reference: Ederer Topical Report EDR-1, Section C.2.b

Requirement The crane will be designed such that the maximum load motion following a drive train failure is less than one foot and the maximum kinetic energy of the load is less than that resulting from one inch of free fall of the maximum critical load.

Finding: This requirement was implemented. The Ederer hoist emergency brake required a minimum height of 9 inches for actuation. Procedure GM-OI-HE-2, Steps 6.18 and 6.19 were added under Condition Report #200601230 to establish a minimum lift height of 12 inches over objects being traversed.

Documents Reviewed: Generic Licensing Topical Report EDR-1, "Ederer's Nuclear Safety Related Extra Safety And Monitoring (X-SAM) Cranes," Appendix C Supplement, Revision 1
Procedure GM-OI-HE-2, "Auxiliary Building Crane Normal Operation," Revision 15
Condition Report #200601230

Category: Heavy Loads **Topic:** Safe Load Paths

Reference: NUREG 0612, Section 5.1.1

Requirement Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and the spent fuel pool or to impact safe shutdown equipment.

Finding: This requirement was implemented. Procedure GM-OI-HE-0002, Step 6.23 and Attachment 9.4 described the general safe load path between the spent fuel pool, decontamination area and railroad siding. The safe load path was identified with steel plates on the fuel handling floor, stenciled with laser targets. Lasers were provided on the lift yoke and trolley. Lift height reference points were stenciled on the lift yoke.

Cameras were located at the east end of the spent fuel pool, on the west side of the decontamination area, at the south end of the new fuel vault, and on the north and south sides of the railroad siding. Backup cameras were provided in all areas except for the railroad siding, and all cameras had a battery pack backup. The cameras provided displays on two screens at the crane remote operating station. During pre-operational testing, the cameras, lasers, and laser targets were used to ensure the load was carried along the safe load path.

Documents Reviewed: Procedure GM-OI-HE-0002, "Auxiliary Building Crane Normal Operation," Revision 15

Category: Heavy Loads **Topic:** Seismic Restraints

Reference: CoC 1004, Tech Spec 1.2.16

Requirement Seismic restraints shall be provided in the spent fuel building to prevent overturning of a loaded transfer cask if the horizontal acceleration at the transfer cask center of gravity is 0.40g or greater. Determination of the horizontal acceleration acting at the center of the loaded transfer cask must be based on the site peak horizontal ground acceleration, but shall not exceed 0.25g.

Finding: This requirement was implemented. The Stone and Webster calculation determined that the acceleration values at the center of gravity on a loaded transfer cask were between 0.25g and 0.32g. Since these values were less than the 0.40g limit, seismic restraints were not required. The results of the calculation was incorporated into Attachment A, Section 1.2.16 of the 10 CFR 72.212 Report.

Documents Reviewed: Stone and Webster Calculation #SC-3
10 CFR 72.212 Report, Revision Prelim.

Category: Heavy Loads **Topic:** Transfer Cask Alignment

Reference: CoC 1004, Tech Spec 1.2.9

Requirement Prior to canister insertion or retrieval, the transfer cask must be aligned to the Horizontal Storage Module (HSM) such that the longitudinal centerline of the canister is within 1/8 inch of its true position.

Finding: This requirement was implemented. Procedure RE-RR-DFS-0004, Step 7.3 directed the final alignment of the transfer cask to the HSM. The maximum vertical and horizontal tolerances to the longitudinal centerline of the HSM were 1/16 inch. During pre-operational testing, the required vertical, horizontal and longitudinal alignment was achieved using three survey transits and positioning targets attached to the transfer cask trunnions, transfer cask end bell, and HSM.

Documents Reviewed: Procedure RE-RR-DFS-0004, "DSC From TC To HSM Transfer Operations," Revision 1

Category: Heavy Loads **Topic:** Transfer Cask Drop

Reference: CoC 1004, Tech Spec 1.2.10

Requirement In the event of a drop of a loaded transfer cask from a height greater than 15 inches the fuel in the canister shall be returned to the spent fuel pool, the canister shall be removed from service, and the transfer cask shall be inspected for damage. The canister shall not be returned to service until a determination is made that it will continue to provide confinement. The transfer cask shall not be returned to service until a determination is made that it will continue to provide its design functions of transfer and shielding.

Finding: This requirement was implemented. Procedure RE-RR-DFS-0003, Step 5.14.1.B contained the actions to be taken on a drop of a loaded transfer cask from a height greater than 15". The actions were consistent with Technical Specification 1.2.10.

Documents Reviewed: Procedure RE-RR-DFS-0003, "Loaded DSC/TC From Auxiliary Building to ISFSI Operations," Revision 1

Category: Heavy Loads **Topic:** Transfer Cask Operations in Direct Sunlight

Reference: CoC 1004, Tech Spec 1.2.14

Requirement Transfer operations shall not be conducted when the transfer cask is exposed to direct sunlight at an ambient temperature of 100 degrees F or greater. For transfer operations at temperatures of 100 degrees F or greater, a solar shield shall be used to protect the transfer cask from direct sunlight.

Finding: This requirement was implemented. Procedure RE-RR-DFS-0003, Step 5.14.3 provided the restrictions for transfer operations at high temperatures and in direct sunlight. The restrictions were consistent with Technical Specification 1.2.14.

Documents Reviewed: Procedure RE-RR-DFS-0003, "Loaded DSC/TC From Auxiliary Building to ISFSI Operations," Revision 1

Category: Heavy Loads **Topic:** Use of Slings

Reference: ASME B30.9, Section 9-6.10.4

Requirement Slings shall be: a) hitched in a manner that prevents slippage and provides control of the load; b) applied to the bowl of the hook; and c) protected from sharp edges with the use of softeners. Slings shall NOT be: a) constricted, bunched, pinched, twisted or shock loaded; or b) shortened or adjusted by any means not specifically approved by the manufacturer.

Finding: This requirement was implemented. Procedure FCSG-15-25, Step 9.1 provided requirements for the hitch configurations and use of slings. The requirements were consistent with the ASME code. The shielding bell weighed 26.63 tons and was lifted using four endless synthetic round slings. Each sling was attached to two eye-to-eye synthetic round slings in a basket hitch. Softeners were used on the edges of the shielding bell.

The transfer trailer inner shield weighed 27,600 pounds and the outer shield weighed 31,000 pounds. Both were lifted using four endless synthetic web slings. Each sling was attached directly to the shield with swivel "D" rings. All sling certifications were current and the slings were rigged and operated in accordance with Procedure FCSG-15-25.

Documents Reviewed: Procedure FCSG-15-25, "Cranes, Derricks, Rigging and Hoists," Revision 1

Category: Lift Yoke and Trunnions **Topic:** Initial Testing - Lift Yoke

Reference: ANSI N14.6, Sections 6.2.1 / 6.5 / 7.3.1

Requirement Prior to initial use, the lift yoke shall be subjected to a load test. If a single component failure could result in a load drop of a critical load, the yoke shall be tested to 300% of the maximum service load. If two component failures are necessary for a load drop to occur, each component in the load path shall be tested to 150% of the load. Following a ten minute hold, the critical areas and load bearing welds shall be subjected to non destructive testing using the liquid penetrant or magnetic particle methods.

Finding: This requirement was implemented. The lift yoke assembly consisted of a lifting beam with a lifting hook on each end. The lifting beam was connected to the Auxiliary Building crane main load block with a 7" diameter steel pin. The lifting hooks engaged the transfer cask trunnions. Since the lift yoke lifting beam and hooks did not provide redundant load drop protection, a 300% load test was required. The lift yoke was rated at 75 tons (150,000 pounds), which required a load test at a minimum of 450,000 lbs. The load test specification was 520,000 (+ 6000/- 0 lbs) and the load test was conducted at 524,266 lbs. The load was held for 10 minutes.

Redundant drop protection for the lift yoke was provided by two adaptor plate assemblies. Each assembly consisted of an adaptor plate which attached to the crane load block sister hook at the top. Two studs extended down from the adaptor plate to below the bottom of the lift yoke lifting beam. A lower steel plate was inserted under the lifting beam and attached to the studs with nuts. If the main pin failed, a load drop would be prevented by the adaptor plate assemblies and the sister hooks. The adaptor plate assemblies each required a 150% load test at a minimum of 225,000 lbs. The load test specifications for both were 260,000 (+ 6000/- 0 lbs) and both load tests were conducted at 262,203 lbs. The load was held for 10 minutes in both tests.

Following load testing, the lifting beam, lifting hooks, 7" diameter main pin, adaptor plates, studs, and lower steel plates were non destructively tested using the liquid penetrant method. The lifting beam and lifting hooks were subjected to additional non destructive testing using the magnetic particle method. No indications of cracking or permanent deformation were identified.

Documents Reviewed: Ranor, Inc. NUHOMS OS197L On-Site Transfer Cask 75 Ton Lifting Yoke Load Test Report, dated January 29, 2006.
Ranor, Inc. Inspection/Nondestructive Examination Record, dated February 28, 2006.

Category: Lift Yoke and Trunnions **Topic:** Initial Testing - Trunnions

Reference: ANSI N14.6, Sections 6.2.1 / 6.5 / 7.3.1

Requirement Prior to initial use, each trunnion shall be subjected to a test load equal to 150% of the maximum service load OR both trunnions shall be subjected to a test load equal to 300% of the maximum service load. After sustaining the load for a period of not less than ten minutes, critical areas, including load bearing welds, shall be subject to non destructive testing using the liquid penetrant or magnetic particle methods.

Finding: This requirement was implemented. Trunnion load testing was performed by Hitachi Zosen on October 14, 2005 under Proof Load Test Procedure No. 034-T-PLT. Prior to load testing, a liquid penetrant examination of the trunnions and trunnion to shell weldments was performed and documented in examination record VPT-CO6. No indications of cracking or deformation were identified. The testing fixture consisted of a main beam with a lifting arm on each end. A steel plate was installed on top of the transfer cask. Four hydraulic jacks were placed between the main beam and the steel plate. The two lifting arms were then engaged to the trunnions and the jacks were raised to take up the slack. The test specification required the trunnions to be loaded to between 750,000 and 772,500 pounds. The actual loading was 767,273 pounds and was held for ten minutes. Following load testing, liquid penetrant examinations of the trunnions and trunnion-to-shell weldments were performed, as documented in examination record PT-PLT. No indications of cracking or deformation were identified.

Documents Reviewed: Proof Load Test Procedure No. 034-T-PLT, Revision 1
Record of Visual Weld/Liquid Penetrant Examination VPT-C-O6, dated October 12, 2005
Record of Visual Weld/Liquid Penetrant Examination PT-PLT, dated October 14, 2005

Category: Lift Yoke and Trunnions **Topic:** Nil Ductility Transition Temperature Testing

Reference: ANSI N14.6, Section 4.2.6

Requirement Ferritic materials used for load bearing members shall be subjected to a drop weight test in accordance with ASTM E20884 or a Charpy impact test in accordance with ASTM A 370-77. The nil ductility transition temperature (NDTT), as determined by the drop weight test, shall be at least 40 degrees F (22 degrees C) below the anticipated minimum service temperature. Charpy tests shall meet the energy and expansion requirements of the design specification.

Finding: This requirement was implemented. Three test specimens from the materials used for fabricating the lift yoke were sent to Laboratory Testing Inc., for Charpy V-Notch impact testing. The testing was performed at 0 degrees F. The lateral expansion and percent shear were within the limits of the design specification.

The minimum service temperature requirement of 40 degrees F is met through Procedure GM-OI-HE-2 Steps 6.31 and 7.11, which require a minimum temperature of 50 degrees F at the bridge area prior to operating the crane.

Documents Reviewed: Laboratory Testing Inc., Certified Test Report CPS001-05-05-13832-1, dated June 21, 2005
Procedure GM-OI-HE-2, "Auxiliary Building Crane Normal Operation," Revision 15

Category: Quality Assurance **Topic:** Approved QA Program

Reference: CoC 1004, Tech Spec 1.1.3

Requirement Activities at the ISFSI shall be conducted in accordance with an NRC approved quality assurance program that satisfies the applicable requirements of 10 CFR Part 50, Appendix B.

Finding: This requirement was implemented. The licensee notified the NRC on March 29, 2004 of their intent to apply their previously approved Fort Calhoun Station 10 CFR Part 50 Quality Assurance Program to ISFSI activities. Appendix A of the Fort Calhoun Station

USAR was revised to incorporate the ISFSI into the Site Quality Assurance Program. Procedure QAP-1.3 was revised to expand the Quality Assurance Program boundary to include the ISFSI. Procedure QAP-11.6 was developed specifically for applying the Fort Calhoun Station 10 CFR Part 50 Quality Assurance Program to the ISFSI.

Documents Reviewed: OPPD Letter #LIC-04-0042 to the NRC dated March 29, 2004 (ML040920248)
Fort Calhoun Updated Safety Analysis Report, Appendix A, "Quality Assurance Program", Revision 18
Procedure QAP-1.3, "Quality Assurance Program Boundary," Revision 7
Procedure QAP-11.6, "Dry Fuel Storage," Revision 0

Category: Quality Assurance **Topic:** Audits, Assessments, and Surveillances

Reference: 10 CFR 72.176

Requirement The licensee shall carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the Quality Assurance (QA) Program and to determine the effectiveness of the program.

Finding: This requirement was implemented. Procedure QAP-10.1, Section 4.2 established the Fort Calhoun Station formal audit program and schedule. Step 4.2.6 of the procedure permitted supplemental QA surveillances whenever a systematic independent assessment of program effectiveness was considered necessary. Startup and operation of the new dry fuel storage system met the criteria for supplemental surveillances.

Two Quality Surveillance Observations were made at Hitachi Zosen in December of 2004 to determine their fabrication readiness. No issues were identified. One Quality Surveillance Observation was made at Bayshore Concrete in April, 2005 to inspect the completed HSMs prior to shipment. No issues were identified. Nine inspections of the HSM assembly operations were conducted on-site between July 8 and October 10, 2005 and were documented in three Quality Surveillance Observations. One surveillance observation rated the assembly operation as unacceptable. The HSM horizontal centerlines had been incorrectly scribed and drilled during assembly. The centerline markings were subsequently lowered and the alignment targets were relocated. This resolved the unacceptable condition.

A self assessment of dry fuel storage readiness was conducted on-site between February 27 and March 3, 2006. The team included peer evaluators from other stations with ISFSI operational experience. The assessment identified several actions needed in order to be ready for dry fuel storage operations. These actions were subsequently completed.

Documents Reviewed: Procedure QAP-10.1, "Audit Program and Audits," Revision 14
Quality Surveillance Observations #958 and #990 conducted on December 7 and December 11, 2004 at Hitachi Zosen
Quality Surveillance Observation #1304 conducted on April 20, 2005 at Bayshore Concrete.
Quality Surveillance Observations #1483, 1484 and 1556 conducted on July 19, July 22, and October 4, 2005 at the ISFSI pad
Self Assessment SA-06-24 conducted between February 27 and March 3, 2006

Category: Quality Assurance **Topic:** Control of Measuring and Test Equipment

Reference: 10 CFR 72.164

Requirement The licensee shall establish measures to ensure that tools, gauges, instruments and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specific periods to maintain accuracy within necessary limits.

Finding: During the welding and fluid operations pre-operational testing on January 30 through February 2, 2006, this requirement was not fully implemented. Procedure RE-RR-DFS-0002, Step 7.5.18 required a post-test calibration check on the vacuum instruments used to confirm that Technical Specification 1.2.2 for canister dryness was met. Step 7.6.22 required a post-test calibration check on the compound pressure gauge used to confirm that Technical Specification 1.2.3.a for canister cover gas pressure was met. However, a method for performing post testing calibration checks on these instruments was not provided.

Subsequent to the welding demonstration, this issue was satisfactorily resolved. Step 7.5.26 and Attachment 13 of Procedure RE-RR-DFS-0002 were revised to perform post calibration checks on the vacuum instruments using a calibrated master. Step 7.6.23 required a post calibration check on the compound gauge, which was controlled under the licensee's Measuring and Test Equipment program. This closes inspection finding 72-054/0601-04 in Inspection Report 050-00285/06-012; 072-00054/06-001.

Documents Reviewed: Procedure RE-RR-DFS-0002, "Dry Shielded Canister Sealing Operations," Revision 1

Category: Quality Assurance **Topic:** Corrective Actions

Reference: 10 CFR 72.172

Requirement The licensee shall establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action taken to preclude repetition. This must be documented and reported to appropriate levels of management.

Finding: This requirement was implemented. Procedure QAP-10.4, Step 4.1 required that conditions adverse to quality be promptly identified and corrected. Nonconformances were corrected under Procedure QAP-7.4. Procedure QAP-10.4, Section 2.0 defined significant conditions adverse to quality as those that warrant an increased level of management attention. These conditions included failures of safety systems, conditions outside the plant design basis, repeat occurrences indicating corrective actions have been ineffective, and gross or widespread non-compliance with the QA Plan. Procedure QAP-10.4, Step 4.2 required that the cause of significant conditions adverse to quality be determined and corrective action taken to preclude repetition. The Condition Review Group determined whether a condition report presented a significant condition adverse to quality. Significant condition reports were assigned to the Condition Review Group. The Plant Review Committee reviewed and approved corrective actions taken to resolve significant conditions adverse to quality.

Documents Reviewed: Procedure QAP-10.4, "Condition Reporting and Corrective Action," Revision 7
Procedure QAP-7.4, "Control of Nonconforming Items," Revision 7

Category: Quality Assurance **Topic:** Handling and Storage Controls

Reference: 10 CFR 72.166

Requirement The licensee shall establish measures to control, in accordance with work and inspection instructions, the handling, storage, and preservation of material and equipment to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere and specific moisture content and temperature levels must be specified and provided.

Finding: This requirement was implemented. Procedure QAP-7.3, Step 4.1.5 classified dry fuel storage structures, systems, and components as Level D. Step 4.3.2 permitted Level D items to be stored outdoors in an area which was well drained, gravel covered or paved, and reasonably removed from construction activities and traffic. Items were required to be stored on cribbing to allow for air circulation and to avoid trapping water. Transnuclear recommended storing the canisters within their original packaging as received from the fabricator, with an additional tarp placed over them to protect them from snow accumulation and seepage into the package. Any openings in the tarp should be well taped to ensure the integrity of the original packaging.

The dry storage canisters were observed to be stored on the ISFSI pad in accordance with Level D storage requirements and in a manner consistent with the vendor's recommendations. The canisters were supported on their wooden shipping cradles to allow for air circulation and to prevent trapping water.

Documents Reviewed: Procedure QAP-7.3, "Storage, Shipping and Handling," Revision 7
E-Mail from TN to Fort Calhoun dated December 9, 2005, "Storage of DSCs"

Category: Quality Assurance **Topic:** Procurement Controls

Reference: 10 CFR 72.154(a)/(b)/(c)

Requirement The licensee shall establish measures to ensure that purchased material, equipment, and services conform to procurement documents. These measures must include provisions for source evaluation and selection, objective evidence of quality furnished by the contractor/subcontractor, inspection at the contractor/subcontractor source and examination of product on delivery. Records shall be available for the life of the ISFSI.

Finding: This requirement was implemented. The helium for canister leak testing and backfilling was supplied by the Linweld Corporation of Waverly, Nebraska. Since Linweld was a commercial grade gas supplier, the licensee elected to perform a commercial grade dedication in order to use the helium at the Fort Calhoun station. On April 10, 2006 licensee personnel traveled to the Linweld facility and observed the filling and sampling of 30 helium gas bottles. The helium purity was verified to be 99.998%. The dedicated helium bottles were tagged and stored in the auxiliary building and were segregated from all other gas bottles.

Procedure NPD-GL 25.0, Section 25.3.3 required ISFSI project personnel to perform receipt inspections of dry fuel storage components. The receipt inspection checklist was provided in Attachment GL-25-01 of the procedure and documented acceptance of the components by the licensee. Procedure NPD-GL 25.0 provided a conditional release option if formal acceptance could not be made. During inspection of the licensee's programs it was determined that spent fuel canister DSC No. 1 had been brought into the auxiliary building for pre-operational testing without formal acceptance or conditional release. Condition Report #200601469 was generated to document this condition. Subsequent to the inspection, the receipt inspection checklist was completed and DSC No. 1 was formally accepted by the licensee.

Documents Reviewed: Commercial Grade Dedication Completion Report No. 21812, dated April 14, 2006
Condition Report #200601469
Procedure NPD-GL 25.0, "Materials Control Management And Receipt At Site Of District Furnished Equipment And Material," Revision 3

Category: Radiation Protection **Topic:** ALARA

Reference: 10 CFR 72.104(b)

Requirement Operational restrictions must be established to meet As Low As Reasonably Achievable (ALARA) objectives for radioactive materials in effluents and direct radiation levels associated with ISFSI operations.

Finding: This requirement was implemented. A comprehensive pre-job briefing was conducted prior to commencing pre-operational testing. The management elements of the briefing included the scope of the dry run, the roles and responsibilities of the shift leads and the oversight function provided by management. The acting shift manager attended the briefing. The industrial safety elements of the briefing included stop work authority, fall protection, ladder and scissor lift safety, personnel protective equipment, tools and housekeeping. The radiological safety elements consisted of postings and access control measures, dose rate estimates and the use of personnel monitoring devices and low dose waiting areas. Due to the expected high dose rates, movements of the loaded transfer cask between the spent fuel pool, decontamination area and transfer trailer will be performed remotely using cameras, lasers and laser targets.

An Electronic Alarming Dosimeter (EAD) with an upper limit of 1,600 rem/hour was placed in the decontamination area approximately three feet from the transfer cask. The EAD reading was displayed on a large digital scoreboard mounted next to it and could be seen remotely with the cameras. The reading also displayed at the Control Monitoring System terminals in room 68 and at the Radiation Protection checkpoint. Radiation Protection personnel were prepared to take surveys with high range instruments, including the DT-375, RO-20, T-Pole, Amp-100, and 3090(-3). These instruments had maximum beta/gamma ranges of 4 to 1000 rem/hour. A Rem Ball ASP-1 was also in use with a maximum range of 120 rem/hour neutron.

Procedure RPI-16 limited personnel exposure through the use of temporary shielding, control of access to high radiation areas, continuous air monitoring during fuel movement, and neutron monitoring. Procedure RPI-16 minimized contamination through use of a sprinkler system inside the shield bell, use of a surfactant on the transfer cask exterior, confirming annulus seal effectiveness through a smear survey, performing contamination surveys of the haul path following loaded transfer trailer movement to the ISFSI, and decontamination of the transfer cask interior prior to returning it to the auxiliary building.

Documents Reviewed: Procedure RPI-16, "Dry Cask Spent Fuel Storage," Revision 1

Category: Radiation Protection **Topic:** Canister Gas Sampling

Reference: CoC 1004, Tech Spec 1.1.2

Requirement If fuel needs to be removed from a canister, precautions must be taken to prevent radiological exposure to personnel from damaged or oxidized fuel. The atmosphere within the canister must be sampled prior to filling the canister with water and removing the inner top cover and shield plugs. If air is present, then appropriate filters should be in place to preclude the uncontrolled release of potential airborne radioactive particulate from the canister. Respirators and supplied air should be considered in accordance with the licensee's radiation protection program.

Finding: This requirement was implemented. Procedure RE-RR-DFS-0007, Steps 7.2.10 through 7.2.20 and Attachment 9.2 were used to obtain the gas sample and to analyze it for fuel cladding damage. Provisions were made for the use of filters to preclude the uncontrolled release of potential airborne radioactive particulate from the canister. The licensee's Radiation Protection program covered the use of respirators and supplied air. The Corporate Health Physicist stated that any sampling of the atmosphere within the canister would be controlled under a Radiation Work Permit specifically generated for that activity.

Documents Reviewed: Procedure RE-RR-DFS-0007, "DSC Lid Removal Operations," Revision 0

Category: Radiation Protection **Topic:** Criticality Monitoring System

Reference: 10 CFR 72.124.c

Requirement A criticality monitoring system shall be maintained in each area where special nuclear material is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Underwater monitoring is not required when special nuclear material is handled or stored beneath water shielding. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration is not required. The NRC has defined "packaged" to begin when the canister lid closure weld is complete.

Finding: This requirement was implemented. Criticality monitoring was provided by two permanently installed area radiation monitors with clearly audible alarms that sounded locally, and in the control room. Area radiation monitor RIA-085 was located outside the south wall of room 68 and area radiation monitor RIA-088 was located outside the south wall of room 69. Both area monitors had a range of 0.7 mrem/hour to 10,000 rem/hour. These area radiation monitors were in service at all times. In addition, Procedure RPI-16, Steps 7.3.1 and 7.13.2 required portable area radiation monitors to be in continuous operation in the spent fuel pool area and in the decontamination area whenever spent fuel was handled.

Documents Reviewed: Procedure RPI-16, "Dry Cask Spent Fuel Storage," Revision 1

Category: Radiation Protection **Topic:** Determining Annulus Seal Effectiveness

Reference: CoC 1004, Tech Spec 1.2.12

Requirement Following placement of each loaded transfer cask into the cask decontamination area, the top region of the canister outer surface and the transfer cask inner surface above the annulus seal shall be decontaminated. Once the annulus seal is removed, a contamination survey of the upper one foot of the canister outer surface shall be taken. The canister smearable surface contamination levels on the outer surface of the canister shall be less than 2,200 disintegrations per minute (dpm) per 100 square centimeters beta-gamma and less than 220 dpm/100 square centimeters alpha.

Finding: This requirement was implemented. Technical Specification 1.2.12 required removable contamination on packages placed in public transport to be within the limits of 49 CFR 173.443. The annulus seal prevented contamination of the canister exterior surface while the transfer cask was underwater in the spent fuel pool. The sequence specified in Technical Specification 1.2.12 was critical to determining the effectiveness of the annulus seal. If the surface area down to one foot below the top of the canister (including the 6 to 8 inches below the seal) was surveyed to be within the contamination limits, the inaccessible surfaces below that level were also considered to be within the contamination limits. The sequence for performing the smear survey initially established in procedure RE-RR-DFS-0002 would not have determined the effectiveness of the annulus seal. Condition Report #200601481 was generated to correct this condition.

Procedure RE-RR-DFS-0002 was revised to provide a sequence that would determine the effectiveness of the annulus seal. Step 7.1.86 deflated and removed the annulus seal. Step 7.1.88 lowered the annulus water level approximately one foot and Step 7.1.89 performed a contamination survey of the upper one foot of the canister (including the 6-8" below the seal). The survey results were documented in Step 7.1.90 with an acceptance criteria of 2,200 dpm per 100 square centimeters beta-gamma and less than 220 dpm per 100 square centimeters alpha.

Documents Reviewed: Procedure RE-RR-DFS-0002, "Dry Shielded Canister Sealing Operations," Revision 1
Condition Report #200601481

Category: Radiation Protection **Topic:** Exposures During Accident Conditions

Reference: 10 CFR 72.106(a)/(b)/(c)

Requirement For each ISFSI, a controlled area must be established. Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident 5 rem Total Effective Dose Equivalent (TEDE) for accident conditions. The minimum distance from the ISFSI to the nearest boundary of the controlled area must be 100 meters. The controlled area may include roads, railroads or waterways as long as arrangements are made to control traffic to protect the public.

Finding: This requirement was implemented. The Fort Calhoun Station controlled area included the Independent Spent Fuel Storage Installation (ISFSI). The controlled area was traversed by the Missouri River and included the exclusion easement boundary on the east side of the river. The nearest distance from the ISFSI to the Missouri River was approximately 480 feet or 146.3 meters. Section 6.2 of the 10 CFR 72.212 Report referred to Section M.11.1.3 and M.11.1.4 of the NUHOMS FSAR which stated that off-normal conditions do not affect the shielding analysis for the NUHOMS system. Therefore, the offsite doses contributed by the ISFSI during operation were the same for off-normal and normal operations. Section M.11.2.5.3 of the FSAR stated that exposure to offsite individuals at a distance of 100 meters would be approximately 42 millirem for the assumed eight hour duration after a drop of the transfer cask and subsequent loss of water from the neutron shield. This was within the 5 rem TEDE limit for accident conditions established in 10 CFR 72.106(b).

The Nebraska Radiological Emergency Response Plan and the Agreement between the State Agencies and OPPD provided that the State of Nebraska would request the U.S. Coast Guard to close the Missouri River in the event of a nuclear plant incident. These arrangements provided for traffic control to protect the public on waterways inside the owner controlled area.

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.
Drawing No. 59058-EY-1A-0, "Independent Spent Fuel Storage Installation Site Location Plan"
FSAR Sections M.11.1.3 and M.11.1.4
Agreement between and among Nebraska Emergency Management Agency and Health and Human Services Regulation and Licensure and Nebraska Public Power District and Omaha Public Power District (OPPD), dated February 1, 2005
State of Nebraska Radiological Emergency Response Plan for Nuclear Power Plant Incidents, dated June 1, 2005

Category: Radiation Protection **Topic:** Horizontal Storage Module Dose Rates

Reference: CoC 1004, Tech Spec 1.2.7.a

Requirement When loaded with a 32PT canister, the Horizontal Storage Module dose rates are limited to 800 mrem/hour on the front surface, 200 mrem/hour on the door centerline and 8 mrem/hour on the end shield wall exterior.

Finding: This requirement was implemented. Following shield door installation, Procedure RE-RR-DFS-0004, Step 7.7.33 required a dose rate survey of the loaded HSM front surface, HSM door centerline and end shield wall exterior. The dose rate limits specified in the procedure were consistent with Technical Specification 1.2.7.a.

Documents Reviewed: Procedure RE-RR-DFS-0004, "DSC From TC To HSM Transfer Operations," Revision 1

Category: Records **Topic:** Notice of Initial Loading

Reference: 10 CFR 72.212(b)(1)(i)

Requirement The general licensee shall notify the NRC at least 90 days prior to first storage of spent fuel.

Finding: This requirement was implemented. The licensee had provided a letter to the NRC dated December 2, 2005 that documented their intent to load irradiated fuel assemblies at Fort Calhoun beginning March 27, 2006 or soon thereafter under a general license.

Documents Reviewed: OPPD Letter to NRC dated December 2, 2005

Category: Records **Topic:** Registration of Casks with NRC

Reference: 10 CFR 72.212(b)(1)(ii)

Requirement The general licensee shall register the use of each cask with the NRC no later than 30 days after using the cask to store spent fuel.

Finding: This requirement was implemented. Procedure RE-RR-DFS-0004, Step 7.7.37 required the Reactor Engineer to notify Licensing to register the cask with the NRC in accordance with 10 CFR 72.212(b)(1)(ii).

Documents Reviewed: Procedure RE-RR-DFS-0004, "DSC From TC to HSM Transfer Operations," Revision 1

Category: Records **Topic:** Retention of 72.212 Analysis

Reference: 10 CFR 72.212(b)(2)(i)

Requirement A copy of the 10 CFR 72.212 analysis shall be retained until spent fuel is no longer stored under the general license issued under 10 CFR 72.210.

Finding: This requirement was implemented. The licensee had revised Procedure SO-C-6 to classify the 72.212 Report and its associated evaluations and screenings as licensing basis documents. The Fort Calhoun licensing basis documents were required to be maintained for the life of the asset, plus 10 years in the OPPD Records Management Records Retention Inventory. Procedure QAP-3.6, Step 4.3.3 required the 10 CFR 72.212 Report to be retained until the ISFSI was no longer in use.

Documents Reviewed: Standing Order SO-C-6, "Conduct of the Fort Calhoun Station On-Site Dry Fuel Storage Program Operations," Revision 0
Procedure QAP-3.6, "10 CFR 72 Licensing/FSAR Control," Revision 0

Category: Records **Topic:** Retention of CoC and Referenced Documents

Reference: 10 CFR 72.212(b)(7)

Requirement The general licensee shall maintain a copy of the Certificate of Compliance (CoC) and the documents referenced in the certificate.

Finding: This requirement was implemented. The licensee planned to use their existing record retention system for controlling the dry fuel storage records. The Fort Calhoun Records Management Records Retention Inventory process included applicable categories and requirements for the retention of the Dry Fuel Storage Certificate of Compliance (CoC), Technical Specifications, Updated Safety Analysis Report (USAR) and NRC Safety Evaluation Report (SER). All records associated with dry fuel storage activities were required to be maintained as long as spent fuel was stored at the ISFSI.

Documents Reviewed: Procedure QAP-3.4, "Records Management," Revision 8
Procedure QAP-3.6, "10 CFR 72 Licensing/FSAR Control," Revision 0
Procedure QAP-11.6, "Dry Fuel Storage," Revision 0

Category: Records **Topic:** Retention of Quality Assurance Records

Reference: 10 CFR 72.174

Requirement The licensee shall maintain sufficient records to furnish evidence of activities affecting quality. The records must include design records; records of use; and the results of reviews, inspections, tests, audits, monitoring of work performance, and materials analysis. The records must include closely related data such as qualifications of personnel, procedures, and equipment. Inspection and test records must identify the inspector/data recorder, type of observation, results, acceptability, and actions taken concerning deficiencies. Records must be maintained until termination of the license.

Finding: This requirement was implemented. Quality Assurance Plan Procedure QAP 3.4 was revised to include the requirement for retention of the 10 CFR Part 72 permanent Quality Assurance records. Requirements included in the revision were retention of records of design, fabrication, erection, testing, modifications to the ISFSI design, inspection records, and personnel training records.

Documents Reviewed: Procedure QAP 3.4, "Records Management," Revision 8

Category: Safety Evaluations **Topic:** Cask Design Changes

Reference: 10 CFR 72.48(c)(1)

Requirement A licensee can make changes to their facility or storage cask design if certain criteria are met as listed in 10 CFR 72.48.

Finding: This requirement was not fully implemented. During inspections at the Fort Calhoun Station and the Transnuclear headquarters, the NRC staff identified concerns with the proposed use of the Transnuclear lightweight transfer cask designated OS197L.

The Transnuclear 10 CFR 72.48 evaluation calculated heat removal from the OS197L transfer cask during transport on the transfer trailer with additional shielding. The NRC staff questioned whether the methodology used for the calculation was appropriate for the transfer trailer configuration. Due to time constraints, the licensee elected to request an exemption from 10 CFR 72.48(c)(2)(viii) rather than engage in further study to definitively answer the question raised. The Fort Calhoun CoC 1004 Exemption Request was received by the NRC on June 9, 2006 and is under review.

If the exemption is approved, the 72.212 Report will be revised by the licensee to incorporate the provisions of the exemption (Attachment 2, Safety Evaluations)

Documents Reviewed: 10 CFR 72.212 Report, Revision Prelim.
Request For Exemption From NUHOMS Certificate of Compliance No. 1004, Amendment No. 8 transmitted to the NRC under OPPD letter LIC-06-0056 dated June 9, 2006

Category: Training **Topic:** Approved Training Program

Reference: 10 CFR 72.44(b)(4)

Requirement The licensee shall have a training program in effect that covers the training and certification of personnel that meet the requirements of Subpart I before the licensee receives spent fuel at the ISFSI.

Finding: This requirement was implemented. The Fort Calhoun 10 CFR Part 50 Training Program used the Instructional Systems Design (ISD) process for developing performance based training. The ISD process was founded in a job and task analysis. The TriVis Corporation had performed a job and task analysis for the ISFSI operations and had used the ISD process for developing the training. However, at the time of the programs inspection, the TriVis training program for ISFSI personnel had not been recognized by, or approved under, the Fort Calhoun Training Program. Condition Report #200601503 was generated to document this condition.

Subsequently, the Training Program Master Plan was upgraded to incorporate ISFSI training into the Fort Calhoun Station training system. Classroom and On-The-Job training requirements were defined and Performance Evaluation Guides were implemented. All personnel engaged in dry fuel storage operations completed the ISFSI Training Program and were certified by the licensee.

Documents Reviewed: Fort Calhoun Station Training Program Master Plan, "Dry Cask Spent Fuel Storage Training Program," Revision 0

Category: Training **Topic:** Documentation

Reference: FSAR 1004, Section 9.3.3

Requirement The licensee's plant training organization is responsible for training programs and for maintaining up-to-date records on the status of personnel training.

Finding: This requirement was implemented. The Fort Calhoun Station Training Program Master Plan required attendance sheets (Form TAP-5D), graded evaluations, and completed task qualification cards to be forwarded to the Engineering Support Personnel (ESP) Training Coordinator. Once the required training was completed, the ESP Training Coordinator forwarded the records to the Training Records Department for entry into the Fort Calhoun System training records. The heavy loads phase of the licensee's pre-operational testing program constituted the final training for ISFSI personnel. At the close of heavy loads testing, the completed training records were scheduled to be reviewed and forwarded to the ESP Training Coordinator.

Documents Reviewed: Fort Calhoun Station Training Program Master Plan, "Dry Cask Spent Fuel Storage Training Program," Revision 0

Category: Training **Topic:** Dry Run Scope

Reference: CoC 1004, Tech Spec 1.1.6

Requirement A dry run of the canister loading, transfer cask handling, and canister insertion into the Horizontal Storage Module (HSM) shall be held. The dry run shall include: 1) functional testing of the transfer cask and lift yoke over the entire route; 2) loading the canister into the transfer cask and installing the annulus seal; 3) transporting the transfer cask to the ISFSI and aligning it with the HSM; 4) inserting a weighted canister into the HSM and retrieving it; 5) loading a mock-up fuel assembly into the canister; 6) canister sealing, vacuum drying and helium backfilling; 7) opening the canister; and 8) returning the canister and transfer cask to the spent fuel pool.

Finding: This requirement was implemented. The dry run of the canister loading, transfer cask handling, and canister insertion into the Horizontal Storage Module (HSM) was completed on May 1 through 5, 2006 at the Fort Calhoun Station. The dry run was conducted under Work Order #226330 and Radiation Work Permit (RWP) 3005, Task 03 and was performed in the following sequence.

The dry run began with the transfer trailer in the railroad siding loaded with the transfer cask containing a weighted canister. The prime mover was used to haul the transfer trailer to the ISFSI pad and to position it in front of the Horizontal Storage Module (HSM). The transfer trailer leveling stands were extended to take the weight off the wheels in order to level the trailer. The transfer trailer support skid was then adjusted to bring the transfer cask into rough alignment with the HSM. This satisfied Technical Specification 1.1.6.3.

Survey transits and positioning targets were used to precisely align the transfer cask to within 1/16" of the longitudinal centerline of the HSM. Once the precision alignment was made, the cask restraint bolts were installed. The hydraulic ram was then extended to push the weighted canister into the HSM. With the ram fully extended, the grapple was disengaged and the ram was withdrawn. The hydraulic ram was then fully re-extended and re-aligned with the canister grapple fitting. Once aligned, the grapple was engaged and the ram was withdrawn to retrieve the canister from the HSM back into the transfer cask. This satisfied Technical Specification 1.1.6.4.

The loaded transfer trailer was then hauled back to the auxiliary building railroad siding. The weighted canister was removed from the transfer cask and the unweighted canister was inserted into the transfer cask. The transfer cask was lifted from the railroad siding and lowered into the decontamination area using the lift yoke. In the decontamination area, the annulus was filled with demineralized water and the annulus seal was installed. This satisfied Technical Specification 1.1.6.2.

The transfer cask was then lifted from the decontamination area and moved to the cask loading area of the spent fuel pool using the lift yoke. This satisfied Technical Specification 1.1.6.8.

The completed dry run sequence functionally tested the transfer cask and lift yoke over the entire route between the spent fuel pool, decontamination area and transfer trailer. This satisfied Technical Specification 1.1.6.1.

In the cask loading area of the spent fuel pool, the dummy fuel assembly was inserted into the canister. This satisfied Technical Specification 1.1.6.5. Handling of the dummy fuel assembly was not observed by the NRC.

Canister sealing, vacuum drying and helium backfilling operations were observed by the NRC on January 30 through February 2, 2006 at the TriVis facility in Pelham, AL. The results of the inspection were documented in Inspection Report 050-000285/06-012; 072-00054/06-001. This satisfied Technical Specification 1.1.6.6.

The licensee did not include opening the canister in their pre-operational testing program. Instead, the licensee took credit for the canister opening demonstration conducted by PCI Energy Services, LLC at the Point Beach station in August 2004. The NRC observed that demonstration and documented the results in Inspection Report 072-00005/2004-001. This was acceptable to the NRC inspector since the Point Beach mock-up canister and canister opening process were similar to the Fort Calhoun Station application. This satisfied Technical Specification 1.1.6.7.

Documents Reviewed: Work Order #226330
Radiation Work Permit (RWP) 3005, Task 03

Category: Training **Topic:** General Training For Station Personnel

Reference: CoC 1004, Tech Spec 1.1.5

Requirement A training module shall be developed for the existing licensee's training program establishing an ISFSI training and certification program. This module shall include an overview of the Standardized NUHOMS design, ISFSI facility design, and Certificate of Compliance conditions. The module shall also include fuel loading, transfer cask handling, canister transfer, and off-normal event procedures.

Finding: This requirement was implemented. Training was provided to the Fort Calhoun Station leaders, Plant Review Committee, and Operations personnel through Lesson Plan NDS01-L001D. The training included an overview of: a) the regulations and standards contained in the SAR, CoC, and General License Conditions; b) the NUHOMS system structural, thermal, confinement, shielding and criticality design criteria; c) the major components of the NUHOMS system and ISFSI; and d) the major steps of the loading and unloading procedures.

Response to off normal events would be directed by the Fort Calhoun shift manager using station procedures. The dry fuel storage procedures directed the loading crew to notify the on-duty shift manager for all off-normal events.

Documents Reviewed: Lesson Plan NDS01-L001D, "Dry Fuel Storage Overview," Revision 0

Category: Training **Topic:** Specific Training For Loading Personnel

Reference: FSAR 1004, Sections 9.3.1.1 and 9.3.1.2

Requirement Generalized training should be provided to ISFSI personnel in the applicable regulations and standards and the engineering principles of passive cooling, radiological shielding and structural characteristics of the ISFSI. Detailed training shall be provided for canister preparation and handling, fuel loading, transfer cask preparation and handling, and transfer trailer loading.

Finding: This requirement was implemented. The TriVis Corporation had provided generalized classroom training for all personnel in: a) the regulations and standards contained in the SAR, CoC, and General License Conditions; b) the NUHOMS system structural, thermal, confinement, shielding and criticality design criteria; c) the major components of the NUHOMS system and ISFSI; and d) the major steps of the loading and unloading procedures.

TriVis had provided detailed classroom training for all ISFSI loading personnel in: a) preparing the transfer cask and canister for fuel loading; b) loading, sealing, drying, and inerting the canister; c) transporting the canister to the ISFSI; d) inserting the canister into the horizontal storage module; e) retrieving the canister from the horizontal storage module; f) transporting the canister back to the Auxiliary Building; g) removing the canister lids; and h) unloading the spent fuel back into the spent fuel pool.

TriVis had provided detailed On-The-Job training for all ISFSI loading personnel in: a) installing and removing the transfer cask top cover and ram access cover; b) installing the top shield plug, inner top cover and outer top cover; c) loading the transfer cask onto, and unloading it from, the transfer trailer; d) inserting the canister into the transfer cask; e) filling and draining the annulus; f) installing and removing the annulus seal; g) connecting and disconnecting the annulus pressurization tank; h) placing the transfer cask into the cask loading area of the spent fuel pool; i) removing the transfer cask from the spent fuel pool and installing it into the shielding sleeve; j) filling and draining the transfer cask neutron shield; k) performing canister vacuum drying and helium backfilling; l) transporting the transfer trailer between the auxiliary building and the ISFSI along the haul route; m) installing and removing the horizontal storage module shield door; n) aligning the transfer cask to the horizontal storage module; o) transferring the canister from the transfer cask into the horizontal storage module; and p) retrieving the canister back into the transfer cask.

The Fort Calhoun Station Training Department provided detailed On-The-Job training for the crane and equipment operators, riggers, fuel handlers, and welders in: a) moving a fuel assembly within the spent fuel pool; b) forklift operation; c) operation of the Auxiliary Building crane; d) filler metal issue and receipt; e) JLG aerial lift operation; and f) rigging practices.

Documents Reviewed: Fort Calhoun Station Training Program Master Plan, "Dry Cask Spent Fuel Storage Training Program," Revision 0
Classroom Lesson Plans NDS01-L001D, NDS01-L002D, NDS02-L001D, NDS02-L002D, and NDS02-L003D
Performance Evaluation Guides DS-01-04, DS-01-05, DS-01-07, DS-01-08, DS-01-09, and DS-01-10
Performance Evaluation Checklists PEC-0475, M8871-P1, M0710-P1, M0712-P1, M8423-P1, M8776-P1, and M8129-P1

Category: Welding and Weld Testing **Topic:** Activities Affecting Quality

Reference: 10 CFR 72.144(b)

Requirement The licensee shall provide control over activities affecting the quality of the systems, structures, and components covered by the Quality Assurance program to an extent commensurate with the approved design of each ISFSI, Monitored Retrievable Storage (MRS), or spent fuel storage cask. The licensee shall ensure that activities affecting quality are accomplished under suitably controlled conditions, such as the use of appropriate equipment.

Finding: During the pre-operational welding demonstration on January 30 through February 2, 2006, this requirement was not fully implemented. Procedure DFS-0002 did not identify the Welding Procedure Specifications (WPS) to be used for each closure weld. Procedure GWS-3 did not provide objective evidence that the Automated Welding System (AWS) welds were made in accordance with the WPS. The calibration requirements for the AWS were not defined.

Subsequent to the welding demonstration, these issues were satisfactorily resolved.

Procedure GWS-3 was revised to include a weld traveler for each closure weld in Attachment 9.3. Each traveler identified the appropriate WPS to be used and provided objective evidence that the weld was made in accordance with the WPS. Transnuclear stated that the Berkeley AWS was designed with an internal diagnostics feature that enabled the machine to maintain its own calibration. This closes inspection finding 72-054/0601-03 in Inspection Report 050-00285/06-012; 072-00054/06-001.

Documents Reviewed: Procedure RE-RR-DFS-0002, "Dry Shielded Canister Sealing Operations," Revision 1
TriVis Procedure GWS-3, "General Welding Standard," Revision 0

Category: Welding and Weld Testing **Topic:** Liquid Penetrant Exam - Contaminants

Reference: ASME Section V, Article 6, T-641

Requirement: The user shall obtain certification of contaminant content for all liquid penetrant materials used on austenitic stainless steels. The certifications shall include the manufacturer's batch number and sample results. Sub-article T-641(b) limits the total halogen (chlorine plus fluorine) content of each agent (penetrant, cleaner and developer) to 1.0 percent by weight when used on austenitic stainless steels.

Finding: This requirement was implemented. The liquid penetrant testing products were supplied by Sherwin Inc., to Fort Calhoun under Purchase Orders 91647 and 92852. The sample analysis for Batch 66-B-56 of the DUBL-CHEK KO-19 cleaner showed a total halogen content of less than 0.00003 percent by weight. The sample analysis for Batch 46-K-54 of the DUBL-CHEK KO-17 penetrant showed a total halogen content of less than 0.00008 percent by weight. The sample analysis for Batch 65-B-71 of the DUBL-CHEK D-100 developer showed a total halogen content of less than 0.001 percent by weight.

Documents Reviewed: Sherwin, Inc. Certifications for DUBL-CHEK Liquid Penetrant Chemicals

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Category: Welding and Weld Testing **Topic:** Liquid Penetrant Exam - High Temperature

Reference: ASME Section V, Article 6, T-653

Requirement When it is not practical to conduct a liquid penetrant examination within the range of 50 to 125 degrees F, the examination procedure at the proposed higher or lower temperature range requires qualification. This shall require the use of a quench cracked aluminum block which in this article is designated as a liquid penetrant comparator block.

Finding: During the pre-operational welding demonstration on January 30 through February 2, 2006, this requirement was not implemented. Procedure QP-9.202, Step 1.4 stated that the liquid penetrant procedure was qualified for use between 60 and 340 degrees F. No documentation was identified to indicate that the procedure had been qualified for high temperature testing, as required by the ASME code.

Subsequent to the welding demonstration, this issue was satisfactorily resolved. TriVis performed liquid penetrant testing on a comparitor block at 340 degrees F using Procedure QP-9.202 and the Sherwin products specified. The indications in the high temperature block were similar to those in the ambient temperature block, thus qualifying the procedure for high temperature use. The high temperature testing was observed and found acceptable by licensee personnel. This closes inspection finding 72-054/0601-01 in Inspection Report 050-00285/06-012; 072-00054/06-001.

Documents Reviewed: Procedure QP-9.202, "Color Contrast Liquid Penetrant (PT) Examination Using the Solvent Removable Method," Revision 0
Leak Testing Specialists, Inc. letter to file dated April 19, 2006

Category: Welding and Weld Testing **Topic:** Liquid Penetrant Exam - Permanent Record

Reference: ASME Section V, Article 6, T-676

Requirement The liquid penetration inspection process, including findings (indications), shall be made a permanent part of the user's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity.

Finding: During the pre-operational welding demonstration on January 30 through February 2, 2006, this requirement was not implemented. Procedure QP-9.202 did not require the use of video, photographic or other means to provide a retrievable record of weld integrity. Neither did the procedure require that findings be documented on the final examination report and entered into a permanent record. The NDE Examination Report in the back of Procedure QP-9.202 did not contain adequate provisions for documenting the nature and location of indications.

Subsequent to the welding demonstration, these issues were satisfactorily resolved. Section 9.1 of Procedure QP-9.202 was revised to require the results of liquid penetrant examinations (including findings) to be documented in Attachments 9.3 and 9.5 of the General Welding Standard, GWS-3. Attachment 9.5 of GWS-3 contained a weld map for documenting rejectable findings. This closes inspection finding 72-054/0601-01 in Inspection Report 050-00285/06-012; 072-00054/06-001.

Documents Reviewed: TriVis Procedure GWS-3, "General Welding Standard," Revision 0
TriVis Procedure QP-9.202, "Color Contrast Liquid Penetrant (PT) Examination Using The Solvent Removable Method," Revision 0

Category: Welding and Weld Testing **Topic:** Visual Testing Procedure Validation

Reference: ASME Section V, Article 9, T-941

Requirement The Visual Testing (VT) procedure shall contain or reference a report of what method was used to demonstrate that the examination procedure was adequate. In general, a fine line 1/32 inch (0.8 mm) or less in width, an artificial imperfection or a simulated condition, located on the surface or a similar surface to that to be examined, may be considered as a method for procedure demonstration. The condition or artificial imperfection should be in the least discernible location on the area surface to be examined to validate the procedure.

Finding: During the pre-operational welding demonstration on January 30 through February 2, 2006, this requirement was not implemented. Procedure QP-9.201 did not contain or reference a report used to validate the procedure. Procedure DFS-0002, Steps 7.2.11 and 7.2.13 allowed use of the Automated Welding System (AWS) camera for visual inspection of tack welds. An additional procedure validation would be required for remote visual testing.

Subsequent to the welding demonstration, these issues were satisfactorily resolved.

Procedure RE-RR-DFS-0002 was revised to remove all references to remote Visual Testing (VT). Remote VT will not be used for inspecting spent fuel canister closure welds at the Fort Calhoun Station. Procedure QP-9.201 was validated for direct visual testing by Leak Testing Specialist, Inc., and Steps 3.6 and 4.2.1 were added to Procedure QP-9.201 to reference the procedure validation. The procedure was validated using a 6 inch scale, with increments of 1/32 inch and 1/64 inch. The scale was placed in the least discernible location on a surface similar to that to be examined. With correct lighting and proper eye position, both increments were clearly visible. This closes inspection finding 72-054/0601-02 in Inspection Report 050-00285/06-012; 072-00054/06-001.

Documents Reviewed: Procedure RE-RR-DFS-0002, "Dry Shielded Canister Sealing Operations", Revision 2
TriVis Procedure QP-9.201, "Visual Weld Examination Of Dry Cask Assembly", Revision 1
Visual Testing Qualification Letter from Leak Testing Specialists, Inc., to file dated June 23, 2006.

Category: Welding and Weld Testing **Topic:** Weld Repairs - Base Metal Defects

Reference: ASME Section III, Article NB-4132

Requirement Weld repairs exceeding in depth the lesser of 3/8 inch (10 mm) or 10 percent of the section thickness, shall be documented on a report which shall include a chart which shows the location and size of the prepared cavity, the welding material identification, the welding procedure, the heat treatment, and the examination results of the weld repair.

Finding: During the pre-operational welding demonstration on January 30 through February 2, 2006, this requirement was not implemented. A major weld repair was made and not documented in accordance with the ASME code. Procedure GWS-3, Step 8.9.3 required weld repairs to be documented in Attachment 9.5 of the procedure. Instead, TriVis documented the weld repair in Attachment 9.3 which did not contain a chart showing the location and size of the prepared cavity.

Subsequent to the welding demonstration, this issue was satisfactorily resolved. Section 9.1 of Procedure QP-9.202 was revised to require the results of liquid penetrant examinations following all weld repairs to be documented in Attachments 9.3 and 9.5 of the General Welding Standard, GWS-3. Attachment 9.3 contained the welding material identification, welding procedure, heat treatment, and examination results (accept/reject). Attachment 9.5 contained a weld map for documenting the location and size of the prepared cavity. This closes inspection finding 72-054/0601-05 in Inspection Report 050-00285/06-012; 072-00054/06-001.

Documents Reviewed: TriVis Procedure GWS-3, "General Welding Standard," Revision 0
TriVis Procedure QP-9.202, "Color Contrast Liquid Penetrant (PT) Examination Using The Solvent Removable Method," Revision 0